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Vol. 2

ORIGINAL

Pilot Program: NRC Severe Reactor Accident Incident Response Training Manual

Severe Reactor Accident Overview

U.S. Nuclear Regulatory Commission

Office of Inspection and Enforcement

T. J. McKenna, J. A. Martin, Jr., C. W. Miller, L. M. Hively,
R. W. Sharpe, J. G. Giitter, R. M. Watkins

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I

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FOREWORD

Over the past few years the Office of Inspection and Enforcement (IE), Division of Emergency Preparedness and Engineering Response, has undertaken a program to upgrade the NRC capabilities to respond to severe reactor accidents. As part of this effort, basic training sessions have been presented by IE staff to all response personnel (Headquarters and regions). Through the process of providing this training a standard student text has evolved.

This pilot training manual has been written to fill the need for a general text on NRC response to reactor accidents. The manual is intended to be the foundation for a course for all NRC response personnel.

This set of manuals is not licensing guidance. Rather, it is designed to present to NRC personnel the best understanding of response planning for a serious reactor accident.

These draft manuals are intended to change over time as NRC staff continues to gain experience. Suggestions are requested and should be sent to the Incident Response Branch.

E L Jordan

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

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PREFACE

Severe Reactor Accident Overview is the second in a series of volumes that collectively summarize the U.S. Nuclear Regulatory Commission (NRC) emergency response during severe power reactor accidents and provide necessary background information. This volume describes elementary perspectives on severe accidents and accident assessment. Other volumes in the series are:

- Volume 1 -- Overview and Summary of Major Points
- Volume 3 -- Response of Licensee and State and Local Officials
- Volume 4 -- Public Protective Actions--Predetermined Criteria and Initial Actions
- Volume 5 -- U.S. Nuclear Regulatory Commission Response

Each volume serves, respectively, as the text for a course of instruction in a series of courses. The volumes have been organized into these training modules to accommodate the scheduling and duty needs of participating NRC staff. Each volume is accompanied by an appendix of slides that can be used to present this material. The slides are called out in the text.

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LIST OF ACRONYMS AND INITIALISMS FOR VOLUMES 1-5

ALARA	As low as reasonably achievable
AMS	Aerial Measurements System(s)
ARAC	Atmospheric Release Advisory Capability
ASC	Administrative Support Coordinator
AST	Administrative Support Team
BT	Base Team (NRC Regional Office)
BWR	Boiling Water Reactor
CDPA	Civil Defense Preparedness Agency
CFA	Cognizant Federal Agency
CFR	Code of Federal Regulations
CL	Congressional Liaison
CRD	Control rod drive
CRDHS	Control rod drive hydraulic system
CSF	Critical Safety Function
DBA	Design Basis Accident
DOC	Department of Commerce, U.S.
DOD	Department of Defense, U.S.
DOE	Department of Energy, U.S.
DOI	Department of Interior, U.S.
DOT	Department of Transportation, U.S.
DSO	Director of Site Operations
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EDO	Executive Director of Operations
ENS	Emergency Notification System
EO	Emergency Officer
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency, U.S.
EPRI	Electrical Power Research Institute
EPZ	Emergency Planning Zone
ERC	Emergency Response Coordinator
ERM	Emergency Response Manager
ERO	Emergency Response Organization
ERT	Emergency Response Team (FEMA organization)
ESF	Engineered Safety Feature
EST	Emergency Support Team (FEMA organization)
ET	NRC Executive Team
ETA	Estimated time of arrival
FBI	Federal Bureau of Investigation
FDA	Food and Drug Administration, U.S.
FEMA	Federal Emergency Management Agency
FRC	Federal Response Center
FRERP	Federal Radiological Emergency Response Plan
FRMAC	Federal Radiological Monitoring and Assessment Center
FRMAP	Federal Radiological Monitoring and Assessment Plan
FTS	Federal Telephone System
GLC	Government Liaison Coordinator
GLM	Government Liaison Manager

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GLO	Government Liaison Officer
GLT	Government Liaison Team
HHS	Health and Human Services, U.S. Department of
HCO	NRC Headquarters Operations Officer
HPCI	High-pressure coolant injection
HPCS	High-pressure core spray
HPN	Health Physics Network
HQ	NRC Headquarters
HUD	Housing and Urban Development, U.S. Department of
IE	NRC Office of Inspection and Enforcement
ICRP	International Commission on Radiological Protection
IDAS	Interactive Dose Assessment System
IRB	Incidence Response Branch
IRC	Incident Response Center
IRDAM	Interactive Rapid Dose Assessment System
JIC	Joint Information Center
LC	Liaison Coordinator
LNO	Liaison Officer
LOCA	Loss of Coolant Accident
LPCI	Low-pressure coolant injection
LPCS	Low-pressure core spray
LT	Liaison Team
LWR	Light Water Reactor
NCS	National Communication System
NMSS	NRC Office of Nuclear Material Safety and Safeguards
NOAA	National Oceanic and Atmospheric Administration
NRC	Nuclear Regulatory Commission, U.S.
NRR	NRC Office of Nuclear Reactor Regulation
NWS	National Weather Service
OC	Operations Center
OSC	Operations Support Center (site)
PA	Public Affairs
PAC	Public Affairs Coordinator
PAG	Protective Action Guides
PAM	Public Affairs Manager
PAR	Protective Action Recommendation
PASS	Post-accident Sampling Systems
PAT	Public Affairs Team
PMC	Protective Measures Coordinator
PMM	Protective Measures Manager
PMT	Protective Measures Team
P-T	Pressure-Temperature
PWR	Pressurized Water Reactor
RA	Regional Administrator
RAT	Radiological Assistance Team
RBE	Relative biological effectiveness
RCIC	Reactor core isolation cooling
RCT	Response Coordination Team
RDO	Regional Duty Officer
R.G.	Regulatory Guide
RHR	Residual heat removal
RI	Resident Inspector
RIRC	Regional Incident Response Center

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RM	Resource Manager
RO	Regional Office
RSC	Reactor Safety Coordinator
RSM	Reactor Safety Manager
RST	Reactor Safety Team
SC	Safeguards/Security Coordinator
SFO	Senior FEMA Official
SGC	Safeguards/Security Coordinator
SGT	Safeguards Team
SI	International System (of measurement)
SLC	Standby liquid control
SM	Safeguards/Security Manager
SO	Status Officer
ST	Site Team
STL	Site Team Leader
TLD	Thermoluminescent dosimeter
TMI-2	Three Mile Island--Unit 2
TSC	Technical Support Center
USDA	U.S. Department of Agriculture
WHO	World Health Organization

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1. SEVERE ACCIDENT OVERVIEW

Slides 1 and 2

1.1 OBJECTIVES

Following completion of this section, the student should be able to describe

- the dose levels required to produce early health effects;
- basic radiation protection criteria, the U.S. Environmental Protection Agency (EPA) and U.S. Food and Drug Administration (FDA) protection action guide (PAG) levels, their relationship to health effects, and how they are used;
- the amount of radioactive material that must be released to produce early health effects and to exceed the Environmental Protection Agency protective action guides at a distance of about 1 mile from the plant;
- the location in the plant of the radioactive material required to produce early health effects, the damage required for its release, the means for detection of this damage, and the level of certainty of the detection;
- conditions in the control room during the initial phase of an accident, including fundamental tasks the control room staff must accomplish;
- how a knowledge of plant conditions (e.g., core temperatures) is essential in projecting consequences of an accident and why in-plant radiation monitors cannot be relied upon entirely to estimate plant releases;

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- the uncertainties associated with dose assessment, including those between (a) what is predicted and what may actually be measured and (b) different model estimates;

- why it may be impossible to project dose adequately during the early phase of a severe accident;

- the role of dose projection during a severe nuclear power plant accident (i.e., General Emergency) and during lesser events; and

- why early protective actions for the population at greatest risk (those nearest the plant) should always be recommended for all directions near the plant and not just downwind.

1.2 THE PROBLEM

Slide 3

Virtually all commercial nuclear power reactors in the United States are either pressurized water reactors (PWRs) or boiling water reactors (BWRs). These types of reactors are called light water reactors (LWRs) because the reactor core is covered with water to allow the nuclear reaction to take place and to keep the core cool (see Appendix D of Vol. 1 for a general description of light water reactors). This manual discusses only accidents at light water reactors.

Another type of reactor, which uses helium gas to cool the core, is called a high-temperature gas-cooled reactor (HTGR). The only high-temperature gas-cooled reactor in the United States is located at Fort St. Vain, Colorado. From a response standpoint, high-temperature gas-cooled reactors have a major advantage over light water reactors. For a high-temperature gas-cooled reactor, it would take several hours after the loss of the systems designed to keep the reactor core cool before a **major** release would be possible (vs minutes for some light water

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reactors).

1.2.1 Radioactive Material Required for Health Effects

Slide 4

The basic sources of safety problems at an operating nuclear power plant are the very large amount of volatile radioactive materials that, if released, could cause off-site health effects and the energy in the core that, if not controlled, could release these fission products. Even if the reactor has shut down, substantial energy is stored in the reactor systems and is being generated by the decay of fission products (decay heat). During the first hour after shutdown, decay heat is on the order of 3% to 5% of full power. For a 3300-MW(t) [3300 megawatts (thermal)] [3,300,000,000 watts (thermal)] reactor, initial decay heat is about 200 MW(t), which, if not controlled, can be a substantial driving force for release of the radioactive materials in the core into the environment.

Slide 5

When discussing quantities of radioactive material, curies are used; the symbol is Ci. A curie is a direct measure of the amount of radioactivity. The number of curies required for various health effects can vary considerably, depending on the type of radiation and how it enters the body (i.e., the pathway). Although curies are helpful in discussing amounts of radioactive material (e.g., curies in the core), they are not useful in discussing the health effects that may result from their release. The amount of radioactivity absorbed by a body is called dose. One unit of dose is the rem. The rem is a measure of potential biological damage induced by radiation exposure to the body, so the rem is used in discussing health effects.

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Scientific notation and units are discussed in Appendixes B and C of Vol. 1.

Slides 6 and 7

Table 1.1 shows the principal components of the 5 billion or so curies of radioactive materials in the core of a light water reactor 30 min after shutdown. Table 1.2 shows inventories of the most volatile radioactive materials (noble gases and radioiodine) in various plant systems. The table also shows the percentage of the various radioactive materials that could be released under worst-accident conditions. Note that the vast majority of radioactive material is contained in the core. All other reactor systems contain less than one-half of 1% of the activity in the core.

Noble gases receive special attention because they are the most likely material to be released during an accident. Radioiodine receives special attention because it can be a major source of the dose to the public early in a severe accident. Radioiodine also can concentrate in the thyroid and in the food chain (i.e., milk). As a result, small quantities of radioiodine can cause damage to the thyroid gland.

Slide 8

As a result of a release of radioactive material to the atmosphere, a person can receive a radiation dose via three major routes (referred to as pathways): (1) shine, (2) inhalation, and (3) ingestion. Individuals receive shine dose externally from direct exposure to a passing cloud (plume) or from contamination deposited on the ground, on themselves, or on other objects. Inhalation dose is received from breathing radioactive material in the plume. Shine and inhalation are generally considered parts of the exposure pathway. Specific protective actions have been established for the plume exposure pathway because of the need for prompt actions. Ingestion dose

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Table 1.1. Summary of volatility of radioactive material in a large [3300-MW(t)] light water reactor core

Volatility	Inventory (millions of Ci)
Very volatile--100% release from fuel possible	
Xenon (Xe) and krypton (Kr)--noble gases	400
Iodine (I)	750
Cesium (Cs)	15
Moderately volatile--1-20% release from fuel possible	
Tellurium (Te)	200
Strontium (Sr) and barium (Br)	370
Not volatile--small percent release possible	
Other	3500

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Table 1.2. Typical inventories of noble gases and iodine in reactor systems

Location	Inventory (Ci)	
	Noble gases (Xe, Kr)	Iodine (I)
Reactor core total	400,000,000	750,000,000
Reactor core gap[a]	30,000,000	14,000,000
Spent fuel storage pool	1,000,000	500,000[b]
Primary coolant[c]	10,000	600[c]
Pressurized water reactor--other systems		
Waste gas storage tank	100,000	1
Boiling water reactor--other systems		
Steam line	10,000[d]	25[d]
Waste gas treatment system	5,000	0.25
Shipping cask	10,000	1

[a]See Sect. 1.2.2 for a discussion of "gap."

[b]One-third of the core is 30 days old; the rest is 1 year old.

[c]Nominal value at normal iodine levels can be much higher or lower (factor of 10) depending on fuel leakage.

[d]Ci/hr (circulating).

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is received from eating or drinking contaminated food or water. Ingestion of milk receives special attention because iodine from a plume can contaminate grass eaten by dairy herds; this radioactive iodine, which can be greatly concentrated in the milk, can then concentrate in the drinker's thyroid gland. These pathways can result in doses being received by various organs. Doses to the whole body and to the thyroid are commonly used as reference levels for protective actions. Cloud shine, ground shine, and inhalation all make major contributions to whole body dose. Inhalation is the major contributor to early thyroid dose.

Slide 9

A release to the atmosphere of major fractions of the radioactive material contained in the reactor core could result in two types of health effects that must be considered in emergency response. The first type is the potential for early deaths and injuries (weeks or months). The second type is the longer-term (latent) health effects (e.g., cancer) that would not be directly observable following an accident. Early injuries generally would appear at doses above 50 to 100 rem to the whole body, and early deaths would be expected at much higher doses (e.g., 200 to 600 rem). It has been estimated that, with only supportive medical treatment, about 50% of the people who receive a whole-body dose of 350 rem would die within 60 days.

Risk of cancer is generally presumed to be proportional to dose, no matter how small. The computer models assume that a collective dose of about 5000 person-rem (e.g., 1 rem to 5000 people) will result in one member of the affected population getting cancer. Because the release is spread over a larger area and therefore over a larger population the farther it moves from the plant, most of the cancers projected by the computer models will result from very small exposures beyond 50 miles from the plant.

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Slide 10

The current dose levels in the Federal guidance that indicate when protective actions are warranted are referred to as protective action guides. These guides have been conservatively established at levels well below the levels that would cause early health effects. At these levels, no health effects would be detectable, even for sensitive populations such as pregnant women.

The Environmental Protection Agency has established protective action guides for plume exposures, and the Food and Drug Administration has established protective action guides for food and agricultural pathways (see Table 1.3). It is important to note that protective action guides are projected doses. At any time, previously incurred doses are not to be considered.

1.2.2 Plant Damage Required for Protective Action Guides to Be Exceeded

Slide 11

It is instructive to compare the amount of radioactive material in light water reactor systems to the amount necessary for release to the atmosphere to induce doses equal to Environmental Protection Agency and Food and Drug Administration protective action guides; note that dose levels ten or more times higher than the protective action guides are required for early injuries or fatalities. Table 1.4 shows the number of curies that would have to be released to the atmosphere to result in doses equal to the protective action guides under meteorological conditions that would, for a given release, produce higher-than-average dose levels at a distance of about 1 mile. Under average meteorological conditions, about ten times more radioactive material would have to be released.

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Table 1.3. Environmental Protection Agency (EPA) and Food and Drug Administration (FDA) protective action guides (PAGs)

Organ	EPA PAGs[a] for plume exposure (rem)	FDA PAGs[b] for food and agricultural products (rem)
Whole body (bone)	1-5	0.5-5
Thyroid	5-25	1.5-15
Other body organs	1-5	0.5-5

[a]At a lower projected dose, protective actions should be considered; at a higher projected dose, protective actions would be warranted.

[b]At a lower projected dose, use of grazing land should be restricted; at a higher projected dose, contaminated milk should be impounded.

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Table 1.4. Atmospheric release (Ci) necessary under poor meteorological conditions[a] to result in protective action guide levels at 1 mile

Radioactive material	Pathway	Curies released[b]	
		15 rem--thyroid[c]	5 rem--whole body
Iodine (I-131)	Milk ingestion	2	-
	Inhalation	600	-
Noble gases (gamma emitters; Xe, Kr)	Cloud shine	-	1,500,000

[a]Conditions that result in doses higher than those projected under average conditions.

[b]Approximate minimum.

[c]Child's thyroid.

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Slides 12 and 13

Table 1.5 shows a summary of annual releases of noble gases and iodine during normal light water reactor operation. It is clear that release rates would have to increase by as much as 100,000 times above normal rates for protective action guides to be exceeded by a 1-hr release.

Slide 14

Comparisons of Tables 1.2 and 1.4 will show that the release of even a very small fraction of the core radioactive material inventory to the atmosphere could result in doses exceeding the protective action guides near the site. However, only the core, spent-fuel storage pool, and the primary coolant contain the requisite amount of the inventory. Accidents not involving those systems (e.g., gas-decay tank rupture) should not result in off-site doses in excess of the Environmental Protection Agency protective action guides.* Furthermore, only the reactor core contains sufficient radioactive material and energy (e.g., decay heat) to result in prompt atmospheric releases that could result in early deaths and injuries off site.

However, even if the core failed, its failure alone would not be sufficient to cause off-site health effects. Other engineered safety features (ESFs) would have to fail also.

---

* One caveat is important: the Food and Drug Administration has proposed a preventative protective action guide of 1.5 rem for the milk pathway; at this level, dairy animals should be removed from likely or actually contaminated pasture. Catastrophic accidental releases of ^{131}I from the waste gas storage tank at a pressurized water reactor site or from the effluent treatment system at a boiling water reactor could result, especially during a period of precipitation, in pasture contamination leading to a projected dose of 1.5 rem or greater via the contaminated milk ingestion pathway.

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Table 1.5. Typical releases

Source	Release	
	Noble gases	Radioiodine
Boiling water reactor		
Annual total, Ci	100,000	1
Release rate to equal annual total, Ci/hr[a]	10	0.0001
Pressurized water reactor		
Annual total, Ci	200	0.02
Release rate to equal annual total, Ci/hr[a]	0.02	0.000002
1-hr release rate to exceed Environmental Protection Agency whole-body protective action guide via cloud shine, Ci/hr	1,500,000	-
1-hr release rate to exceed Food and Drug Administration thyroid dose via inhalation, Ci/hr	-	600

[a]Using 10,000 hr/year.

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Plants were designed to control most releases from these systems. In particular, the containment system is designed to limit releases from the fuel to the atmosphere.

Slides 15 and 16

Four barriers are designed to prevent the release of radioactive materials to the environment from the reactor core (see Fig. 1.1). The first barrier is the structure of the fuel itself in the form of uranium dioxide (UO_2) fuel pellets; UO_2 is a ceramic. The individual fuel pellets are stacked upon each other in zirconium or stainless steel tubing (called fuel pins or cladding) and sealed, forming the second barrier to release. The space between the fuel pins is called the gap. Gaseous radioactive materials produced in the fuel during the fission process are trapped in this gap region. These trapped gases are called the "gap" activity. They are of special importance because they can be released very quickly from the fuel once the cladding is damaged. This would result in a large increase in the levels of radioactivity in the primary coolant at the start of core damage. As seen in Table 1.2, the "gap" contains about 10% of the noble gases and 2% of the iodine in the total core. The fuel rods (pins) are assembled into fuel elements that are all contained within the primary cooling system (vessel and piping), which forms the third barrier to fission product release. Finally, the primary coolant system is contained within the containment building, the fourth barrier.

The term "core damage" is used to indicate major (20% or more) cladding failure (second barrier). A major release is not possible unless a major failure of cladding allows radioactive material to be released from the core into the primary coolant.

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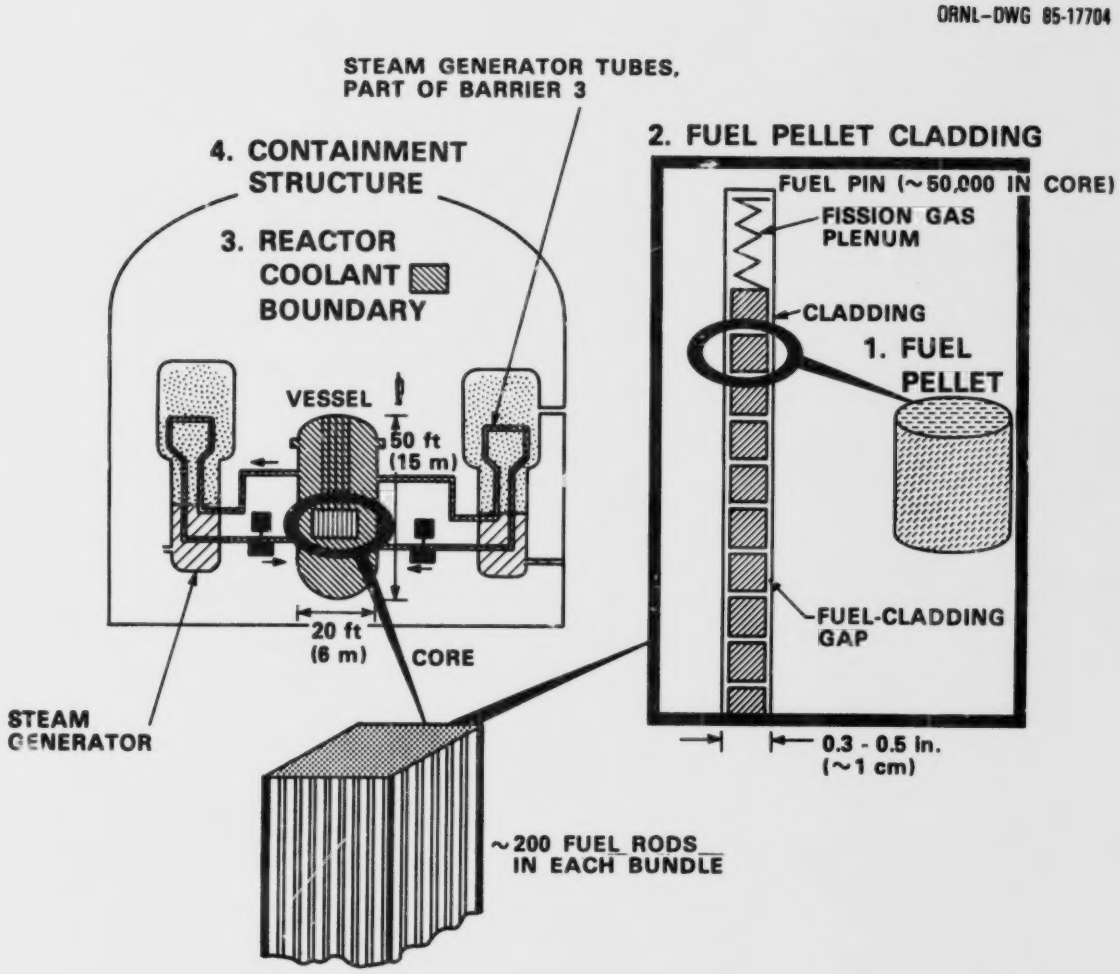

Fig. 1.1. Barriers to release of radioactive material for a pressurized water reactor.

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Slide 17

Core damage (or core melt) should be distinguished from fuel melt. The fuel (UO_2) has the highest melting point [5100°F (2800°C)] of all materials in the core. However, releases of radioactive material from the core that are sufficient to result in major off-site releases are possible at much lower temperatures. At temperatures above 1500°F, fuel cladding failure and release of the "gap" from the core are expected. Once the cladding fails, radioactive material in the fuel can escape into the primary system. At temperatures above 1500°F, the volatile radioactive materials (e.g., iodine and cesium) are being released from the fuel pellets at a considerable rate. Therefore, severe core damage is expected at core temperatures above 1500°F. In a pressurized water reactor, these temperatures could be reached in 15 min if the core were not covered with water. Normally, temperature in the reactor pressure vessel, which houses the core, averages about 600°F. Figure 1.2 illustrates the various levels of core damage, release rates of volatile radioactive materials as temperature increases, and the time after uncovery of the core that these temperatures could be reached in a pressurized water reactor. Reaching these temperatures would take longer in a pressurized water reactor because more of the core must be uncovered before major heat-up starts.

Slide 18

Many systems are designed to protect the fission product barriers. The effectiveness of these systems can be assessed in terms of how well they perform a few critical safety functions

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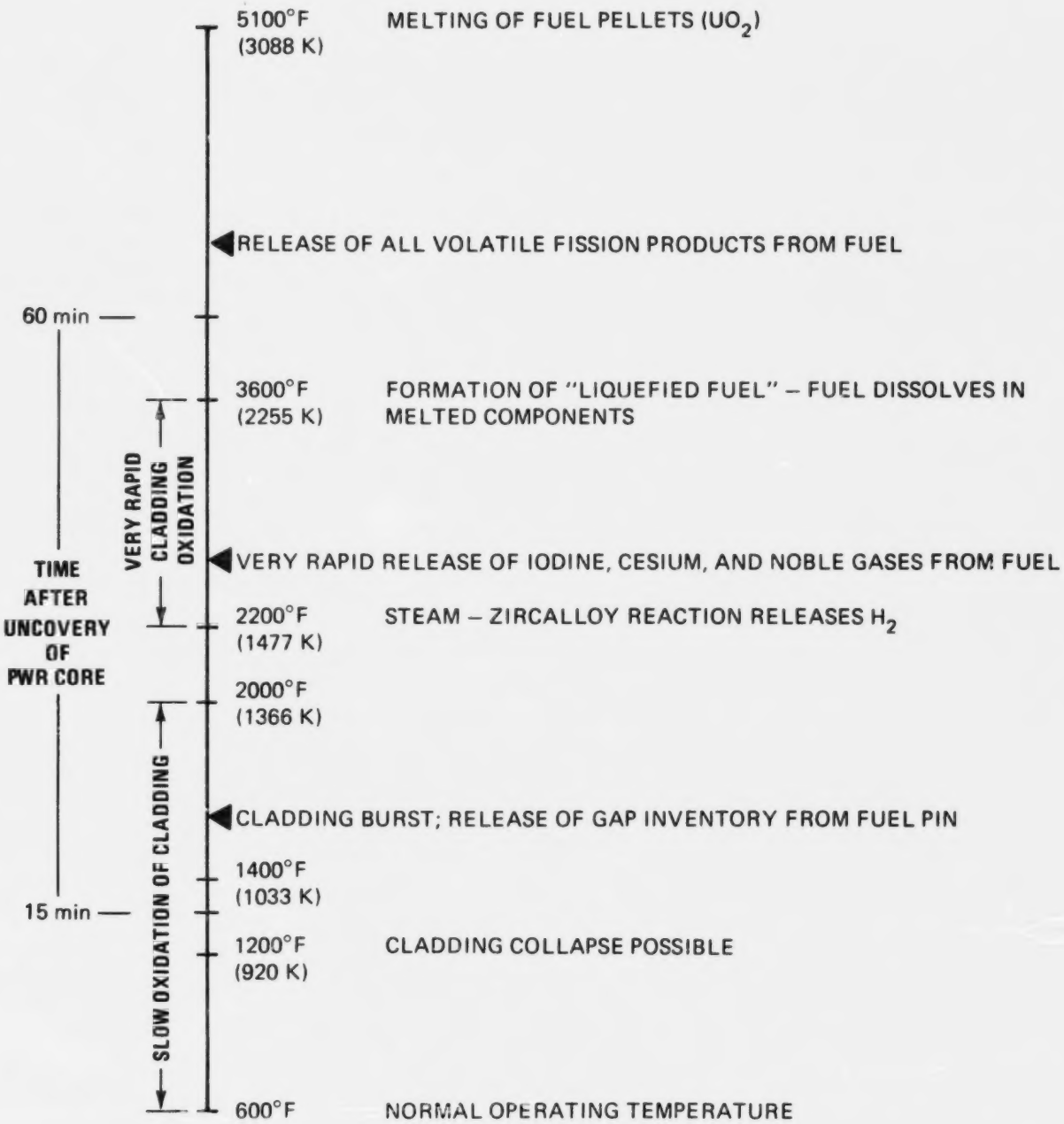

Fig. 1.2. Possible core damage states as core temperature increases.

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(CSFs). The critical safety functions will, if maintained, prevent damage to the core. Each of these functions is performed by a number of redundant engineered safety features. Basically these functions are

1. shut down the reactor (control reactivity),

2. maintain coolant level (keep core covered),

3. provide coolant makeup (keep water flowing through the core), and

4. remove heat to an outside (of containment) sink.

These are the primary functions of the control room staff, whether the situation is an emergency or not. Specific emergency operating procedures (EOPs) and training are provided for this purpose. Therefore, the status of these functions and, consequently, the projected status of the fission product barriers should be known in the control room. The control room should have early warning before a major release.

Slide 19

As an example, two of the critical safety functions are to keep the core covered (provide water) and cool (remove heat). Once the reactor has been shut down, prevention of core damage is a fundamentally simple task. Basically, the task is to keep removing decay heat (energy) from the core by boiling water. If the decay heat is not removed, it will ultimately result in failure of the primary system and possibly in release of fission products to the environment. Only a relatively small flow of water may be required to do this. About 1 hr after shutdown, a flow of only a few hundred gallons of water per minute is needed to keep the core cool. Figure 1.3 shows the amount of water required to remove decay heat by boiling water vs time after

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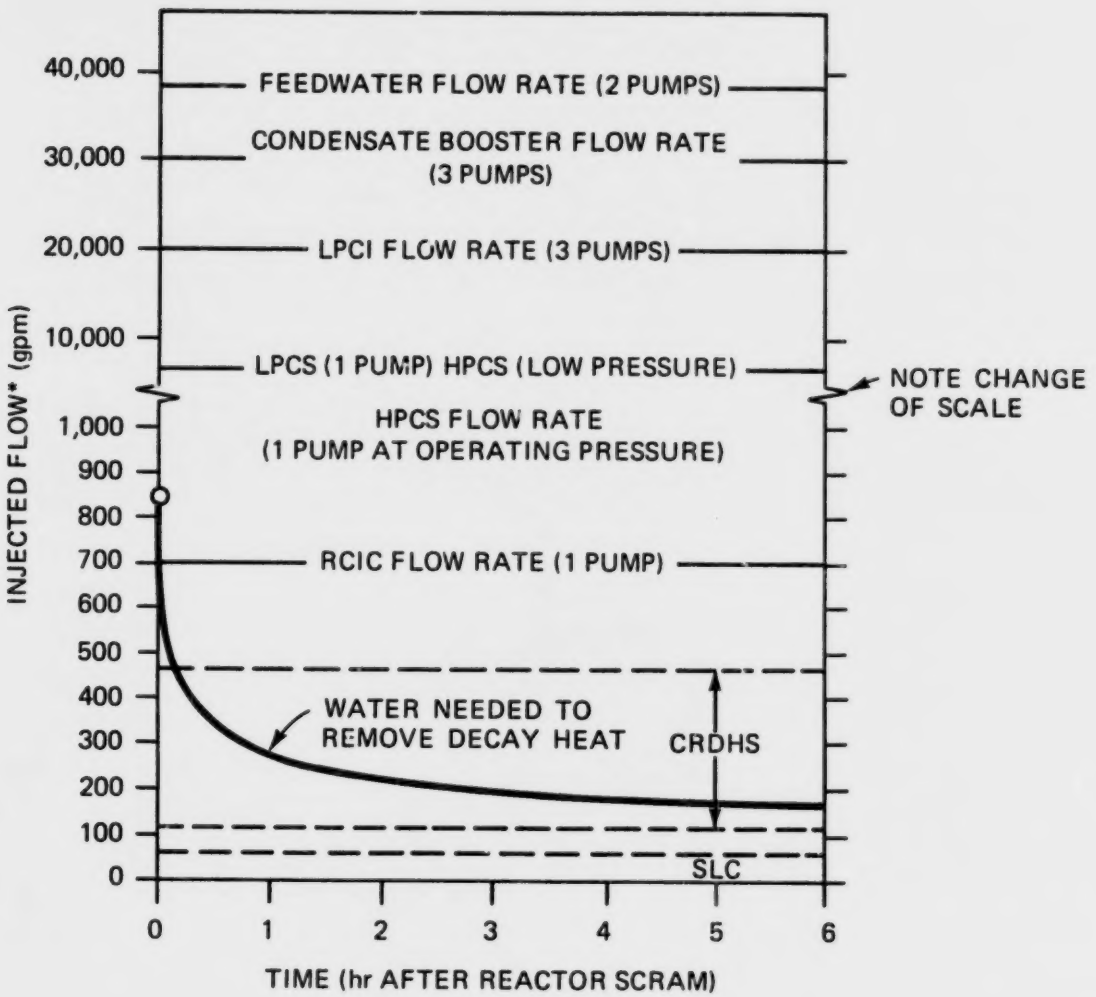

*CALCULATED ASSUMING *CONSTANT REACTOR VESSEL PRESSURE* AND 100% INITIAL POWER DECAY HEAT PER 1979 ANS STD. WITH ACTINIDE DECAY). COOLANT INJECTED AT 305 K (90°F) AND REMOVED AS DRY SATURATED STEAM.

LPCI – LOW-PRESSURE COOLANT INJECTION
RCIC – REACTOR CORE ISOLATION COOLING
LPCS – LOW-PRESSURE CORE SPRAY
HPCS – HIGH-PRESSURE CORE SPRAY
CRDHS – CONTROL ROD DRIVE HYDRAULIC SYSTEM
SLC – STANDBY LIQUID CONTROL

Fig. 1.3. Injected flow required to remove decay heat from the reactor vessel of a 3300-MW(t) plant by boiling water and boiling water reactor systems available to provide the flow.

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shutdown and the boiling water reactor systems available to perform this job.

As illustrated in Fig. 1.3, once the boiling water reactor fission process is essentially terminated by a reactor scram (or trip), decay heat can be removed initially by boiling water at a rate of less than 1000 gal/min. After about 2 hr, boiling water at a rate of 200 gal/min would remove all decay heat. In either case, this would prevent further temperature increases and thereby protect the core.

Figure 1.3 also shows pumping capacities of some safety systems in a boiling water reactor. At high reactor vessel pressure [e.g., 1000 pounds per square inch gage (psig)], only a few systems are normally available to pump water [e.g., the high-pressure core spray (HPCS) system in the figure]. Once the vessel pressure is reduced to below a nominal 300 psig or so (depending on the particular reactor design), high-capacity pumps would be available [e.g., low-pressure core injection (LPCI), feedwater, condensate, and control rod drive hydraulic system (CRDHS) pumps]. Notably, during a severe fire at the Browns Ferry boiling water reactor in 1975, the reactor was cooled by pumping water through the control rod drive cooling system.

Slide 20

Figure 1.4 shows a diagram of water injection systems in a typical boiling water reactor. As can be seen, there are many redundant ways to get sufficient flow to keep the core cool by boiling water and/or to replace coolant lost through a break or leak. Some of these systems are designed to maintain the critical safety functions even under very severe accident conditions such as a total break in the largest piping in the primary reactor coolant system.

Core damage would require failure of several engineered safety features [e.g., the emergency core cooling system (ECCS)] that have been designed to maintain the critical safety

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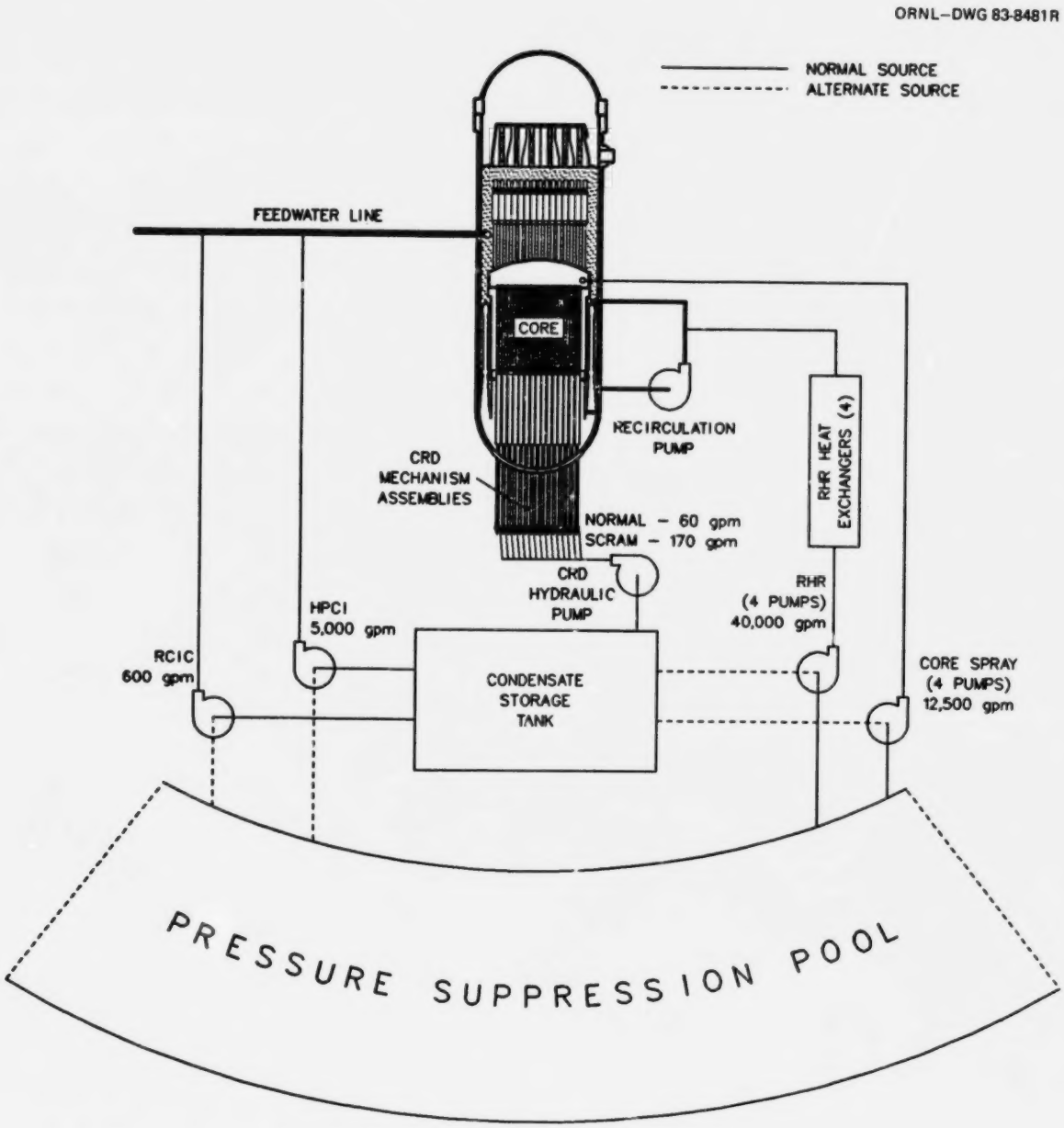

Fig. 1.4. Simplified diagram of typical boiling water reactor vessel water injection sources. (See Appendix D of Vol. 1 for a more detailed discussion of light water reactor systems.)

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functions. Future core damage may be estimated based on the status of these systems. In addition, the extent of core damage may be surmised (grossly) from several key plant safety parameters (e.g., core temperature and coolant level in the reactor vessel and highly elevated levels of radioactivity in the coolant). Although core damage could be readily determined (e.g., high temperatures, high radiation levels), its extent could be very difficult to determine early in an accident sequence. Indicators of the status of key plant safety systems and parameters in conjunction with radiation monitoring provide the best, clearest, and probably only indicators of core damage and movement of radioactive material through plant systems. Clear indications of actual or likely core damage should be evident in the control room <u>before a release</u> to the atmosphere.

Slide 21

Some of the instrumentation and information in the control room that could be used to indicate core damage are listed in Table 1.6.

1.2.3 <u>Plant Conditions Required for a Major Release</u>

Slide 22

In addition to core damage, a major release sufficient to result in early injuries and/or fatalities would require a direct pathway to the environment and a driving force (e.g., steam). In essence, the containment systems would have to fail to perform their intended functions. The radioactive material released from the core must move rapidly through the primary coolant system (third barrier) and containment (fourth barrier) without being significantly filtered or reduced by other methods such as containment sprays. Even if the containment sprays

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Table 1.6. Examples of instrumentation and information available for determination of fuel (core) status

Instrumentation and information	Type of reactor
Immediately available in control room	
Core temperatures--thermocouple	Pressurized water reactor (PWR)
Core-exit temperature levels	Boiling water reactor (BWR)
Containment radiation level	BWR, PWR
Radiation levels from condenser/air ejector	BWR, PWR
Neutron fluxes in core	BWR, PWR
Available after several hours	
Concentration or radiation level in circulating reactor coolant	BWR, PWR
Analysis of primary coolant--gamma spectrum	BWR, PWR
Containment hydrogen level (from samples)	BWR, PWR

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fail, over time natural removal processes (e.g., condensation and scrubbing) will remove most particulate fission products from the atmosphere of an intact containment. Therefore, if the containment holds for several hours and the containment spray systems work or if the release travels through many systems, early injuries or fatalities would be highly unlikely.

Slide 23

On the other hand, severe-accident analyses have identified low-probability accident sequences that could result in significant off-site releases. As can be seen from Fig. 1.5, which summarizes the major sequences, the containment could fail either before or after the release of radioactive material from the core.

Slide 24

Figure 1.6 uses the concept of an event tree to display the potential consequences for public health due to severe accidents. Moving from left to right in the figure, "yes/no" answers to scenarios at the top result in a series of branches, possibly to off-site consequences. For example, if only the radioactive material contained in the fuel pins (gaps) is released with late containment failure, the off-site consequences would be small (branch 7 in Fig. 1.6). If all answers are yes, branch 1 indicates extremely severe off-site consequences. Figures 1.5 and 1.6 illustrate two fundamental public health questions during an emergency response at a light water reactor:

- What is the status of the reactor core?
- What is the status of the reactor containment?

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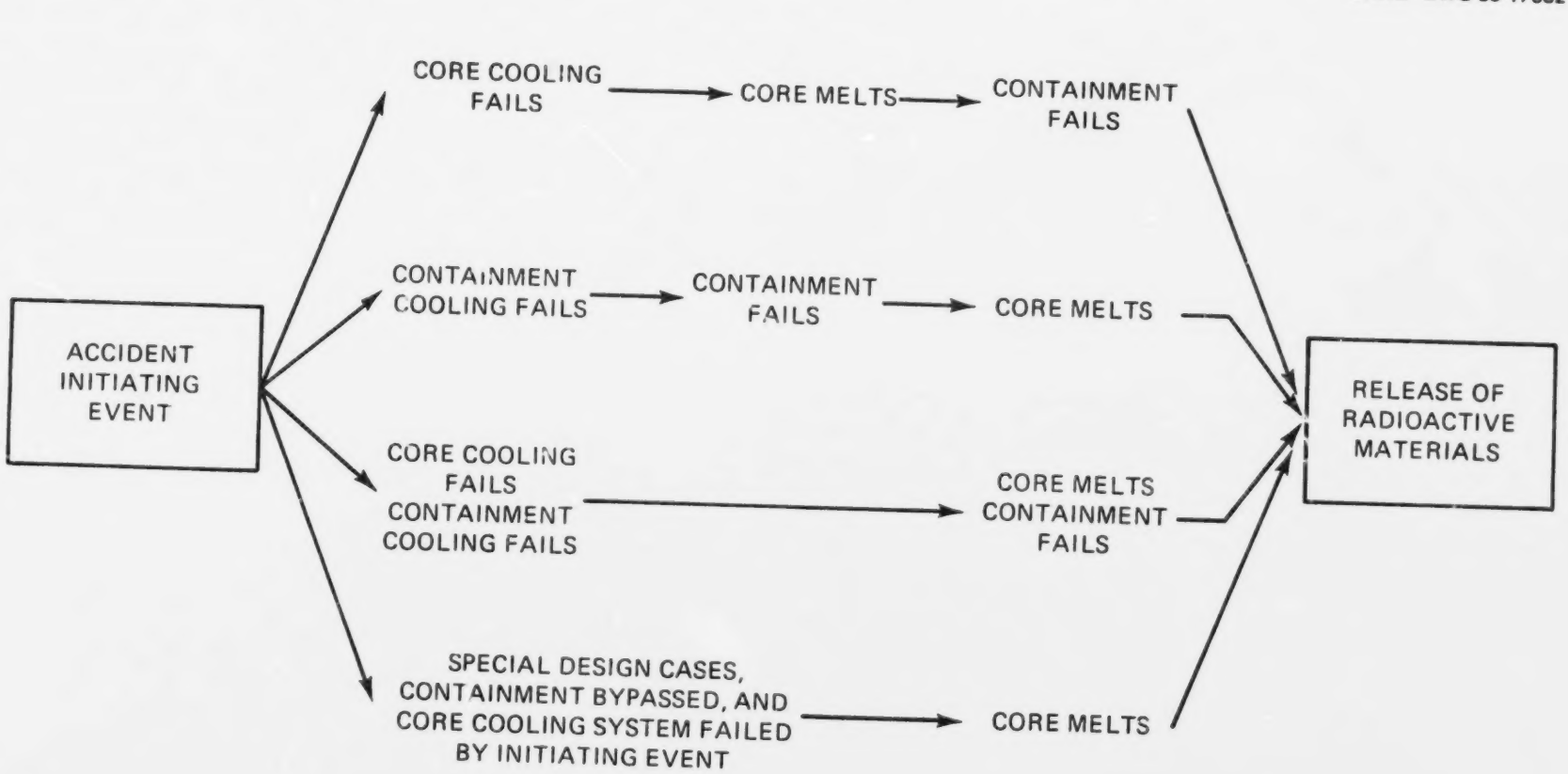

Fig. 1.5. Major sequences in a severe light water reactor accident scenario.

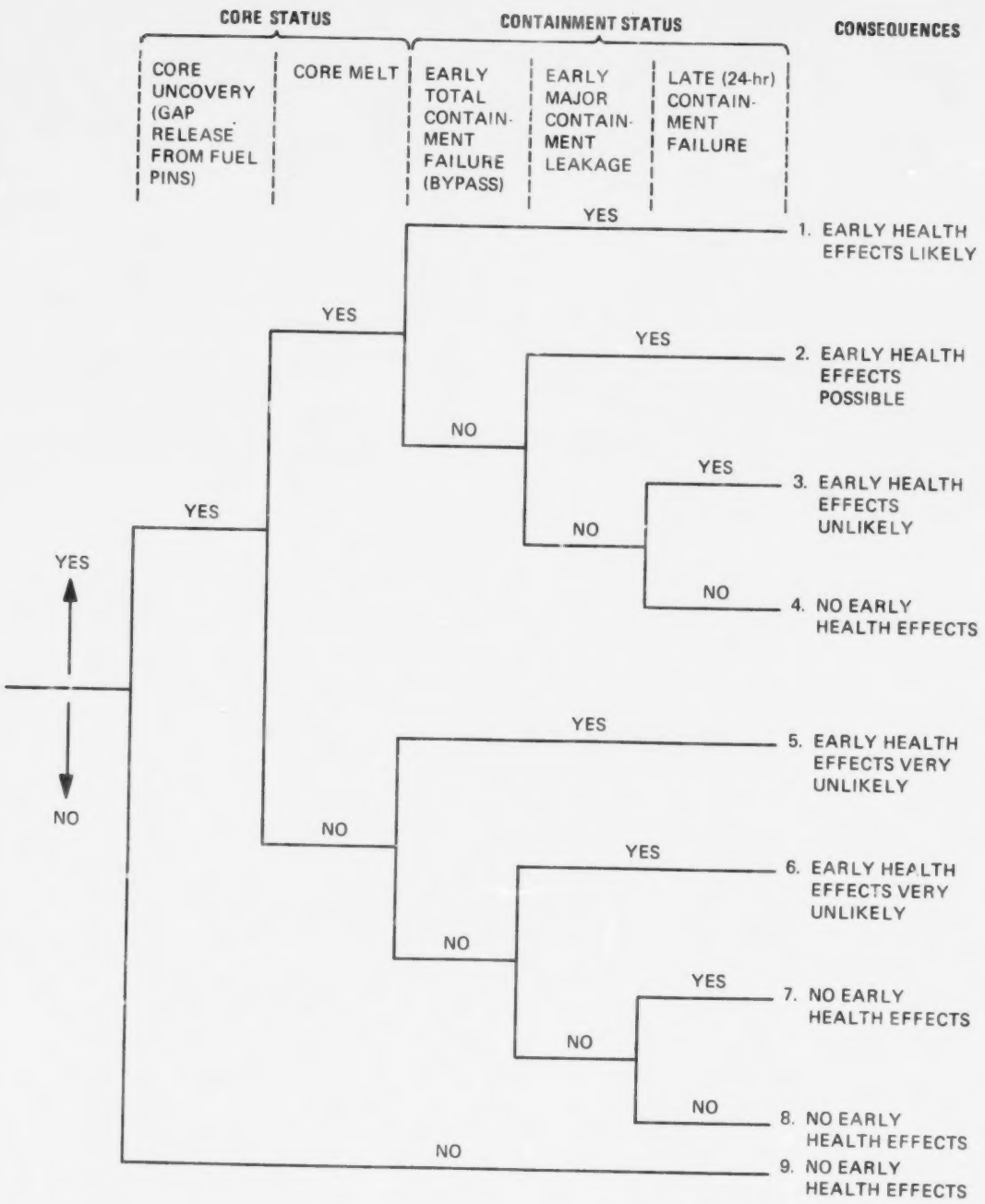
CORE STATUS
CONTAINMENT STATUS
CONSEQUENCES
CORE UNCOVERY (GAP RELEASE FROM FUEL PINS)
CORE MELT
EARLY TOTAL CONTAIN-MENT FAILURE (BYPASS)
EARLY MAJOR CONTAIN-MENT LEAKAGE
LATE (24-hr) CONTAIN-MENT FAILURE
YES
NO
1. EARLY HEALTH EFFECTS LIKELY
2. EARLY HEALTH EFFECTS POSSIBLE
3. EARLY HEALTH EFFECTS UNLIKELY
4. NO EARLY HEALTH EFFECTS
5. EARLY HEALTH EFFECTS VERY UNLIKELY
6. EARLY HEALTH EFFECTS VERY UNLIKELY
7. NO EARLY HEALTH EFFECTS
8. NO EARLY HEALTH EFFECTS
9. NO EARLY HEALTH EFFECTS

Fig. 1.6. Event tree for severe accident consequences.

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The answers to these two questions scope the level of threat to the public and the need for off-site emergency response. This translates to an on-site responsibility for cooling the core and containing any releases from the core.

The control room staff will be able to scope the answer to the first of these questions. As we have seen, there will be considerable indicators of actual core damage. However, depending on the severe accident sequences, the control room staff may not be able to confidently assess containment performance.

Slide 25

The U.S. Nuclear Regulatory Commission (NRC), as part of its severe accident source term work, formed the containment performance working group. This group's findings were summarized in Reassessment of the Technical Bases for Estimating Source Terms (NUREG 0956). The group concluded that, for most risk-dominant accidents, containments would not fail until several hours after core damage and resulting reactor vessel (third fission product barrier) failure. This conclusion was reached partly because containments are not estimated to fail because of overpressure until pressures reach two to three times the design pressures.

Slides 26 and 27

Containments could fail because of overpressurization; Table 1.7 shows the design pressures and estimated failure pressures for several containment types. Actual failure pressure would be unpredictable. Failure could also result from a burning of hydrogen generated during the core damage process. If the valves that are designed to "button up" the containment

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Table 1.7. Typical containment design parameters

Type of plant	Allowable leak rate (vol %/day)	Design pressure (psig)	Estimated failure pressure (psig)
Pressurized water reactor			
Large dry	0.1	47	134
Subatmospheric	0.1	45	119
Ice condenser	0.25	12	50
Boiling water reactor			
Mark I	0.5	62	117
Mark II	0.5	55	140
Mark III	0.4	15	60

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failed to operate correctly, this could also result in a failure (to isolate). Finally, the containment could be bypassed by accidents involving leakage from the primary coolant system directly outside. While static pressures can be monitored and trended during an event, the actual point of failure and the other failure modes would be unpredictable.

1.3 THE CONTROL ROOM ENVIRONMENT DURING AN EMERGENCY

Slide 28

Any NRC (or other outside) respondent must realize that the reactor control room could be a very busy place under accident conditions. First of all, the plant may be well beyond design with many alarms (annunciators) sounding at once. For example, during the Three Mile Island—Unit 2 (TMI-2) accident, many alarms sounded at one time, including all audible alarms on the front control panels. When many alarms sound at once, it is difficult for the reactor operators to separate important, possibly critical, information from ancillary information. Additionally, instruments could be responding erroneously off scale.

The reactor operators will also be faced with an extremely heavy workload, especially during the early stages of any accident. Based on their understanding of the situation, the operators must take immediate actions to assess and correct the problem and bring the accident to a halt. They must continuously monitor the plant's critical safety functions and attempt to restore them to normal and maintain them there. At the same time, the reactor operators are expected to classify the event; activate their emergency organizations; direct damage control; take on-site emergency actions (e.g., site evacuation); and, possibly, make protective action recommendations to off-site officials. In addition, the licensee is expected to notify the NRC and keep them informed of the status of the event and the condition of the plant.

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The NRC staff's contact with the control room staff at such a time must be held to the level necessary to understand plant conditions and must be handled very efficiently so as not to substantially impede plant recovery by the operators. After the plant Technical Support Center (TSC) is established, communications with the plant can be more extensive.

1.4 PREDICTING OFF-SITE CONSEQUENCES

Slide 29

The most important aspect of predicting off-site consequences is to remember that (1) in the absence of core damage, there is little, if any, excessive risk to the public, but (2) given core damage, there must have been major human error or equipment failure. Under these conditions, there may be little assurance that further failures or a major release is not possible because the plant parameters are well beyond their design limits. Some have estimated that as many as one in ten core melt accidents would result in a major release sufficient to cause death and severe injuries off site if effective protective actions were not taken early in the accident sequence.

Slides 30 and 31

In the event of core melt, early, precautionary protective action decisions should be based on in-plant observables (control room indicators) and precalculated dose projections rather than on early, real-time dose projections. This conclusion is based on the relatively high risk of a major release (e.g., 10%) given core damage, the relative ease (of using a few key indicators) for the plant staff to detect major core damage, the large uncertainties associated with projecting containment failure, the great difficulties in making accurate and timely dose projections in the face of the latter

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uncertainty, and the fact that off-site protective actions would be most effective if initiated before a major release occurred (e.g., precautionary evacuation).

1.4.1 Protective Action Initiation Based on Plant Conditions

Slide 32

Once a plant experiences imminent or actual core damage, it would be very difficult or even impossible to predict performance of the containment and other systems that are designed to mitigate the consequences of core damage. Because actions to protect the public must be initiated before or upon a major release to the atmosphere for them to be fully effective, the control room staff must rely on core-condition indicators for recommending off-site action.

Slide 33

A major release will be preceded by a systems failure (start of accident), failure of one or more engineered safety features, failure to meet one or more of the critical safety functions, failure of fission product barriers, and movement of radioactive material through plant systems. Considerable instrumentation exists to indicate the status of the critical safety functions, fission product barriers, and (grossly) the movement of radioactive material.

Slides 34 and 35

As discussed in Vols. 1 and 3, current regulations require nuclear power plants to establish four levels of emergencies (classes) for which various levels of response are preplanned. Licensees have established and incorporated into their procedures emergency action levels (EALs) based on control room instrumentation that would indicate the class of emergency. The

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most serious level of emergency is a General Emergency, which should be declared when plant conditions indicate that immediate protective actions should be taken off site. Severe core-damage events have a very real potential for severe off-site health effects and are to be classified as General Emergencies, which would warrant immediate protective action recommendations off site. While some events have been postulated that could lead immediately (quickly) to core damage, most severe accidents studied would be classified as general emergencies by the emergency action levels well before a major release.

Slides 36 and 37

Figure 1.7 shows an example of the emergency action levels established by a licensee to indicate actual or projected core damage (General Emergency). Case A shows the criteria observable in the control room for declaring a General Emergency before core damage based on loss of a critical safety function (ability to shut down the reactor) and indication (containment temperature and pressure) of conditions that indicate imminent core damage. Case B shows the criteria for declaring a General Emergency based on radiological indications of core damage. In this case, core damage would be indicated based on the containment dome monitor reading or a hand-held instrument dose rate reading outside the containment.

Slides 38 and 39

In addition to the emergency action levels, licensees have been required to develop procedures for assessing the status of the core. These procedures indicate the relationships of various plant instruments (e.g., containment monitor, water level, or thermocouple readings) and degrees of core damage. These relationships must be used with caution and may be considered as only gross indicators of core damage conditions.

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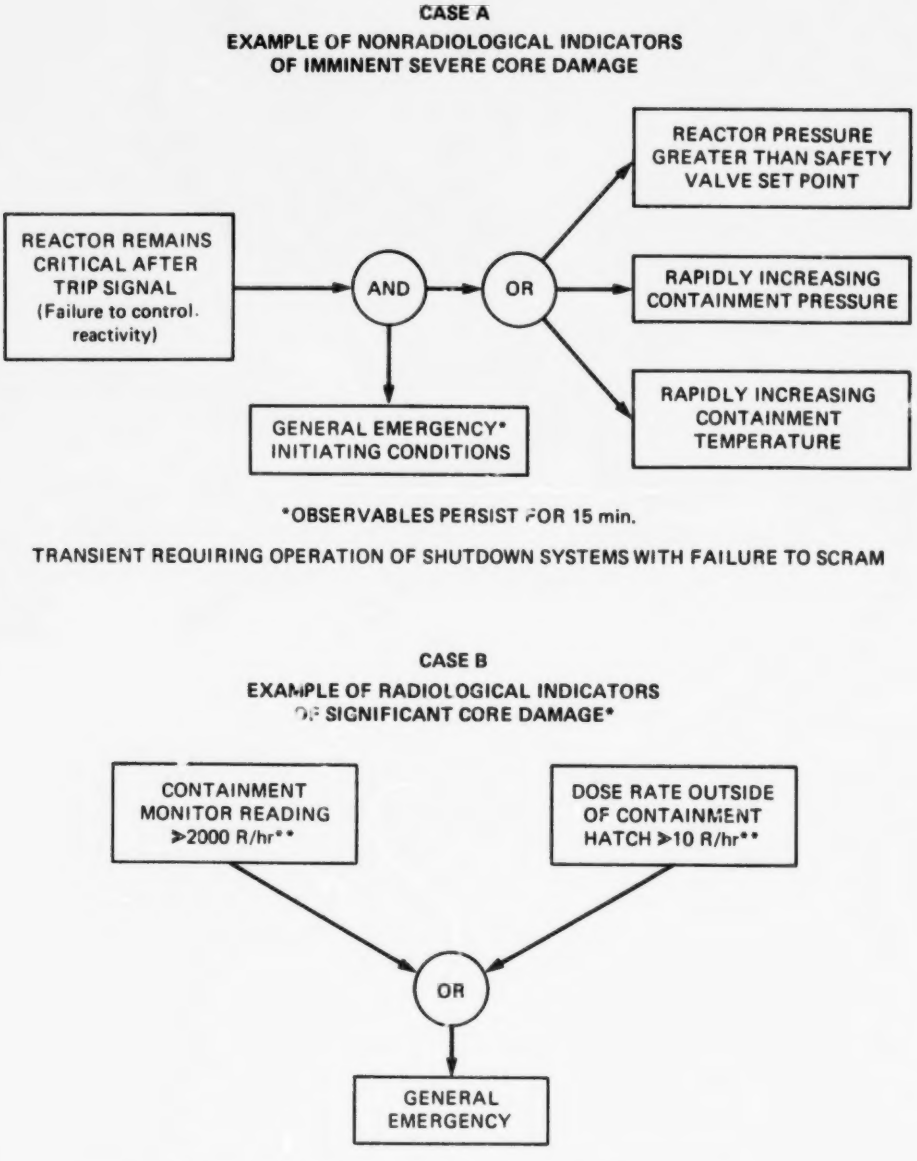
CASE A
EXAMPLE OF NONRADIOLOGICAL INDICATORS
OF IMMINENT SEVERE CORE DAMAGE
REACTOR REMAINS CRITICAL AFTER TRIP SIGNAL (Failure to control reactivity)
AND
OR
REACTOR PRESSURE GREATER THAN SAFETY VALVE SET POINT
RAPIDLY INCREASING CONTAINMENT PRESSURE
RAPIDLY INCREASING CONTAINMENT TEMPERATURE
GENERAL EMERGENCY* INITIATING CONDITIONS
*OBSERVABLES PERSIST FOR 15 min.
TRANSIENT REQUIRING OPERATION OF SHUTDOWN SYSTEMS WITH FAILURE TO SCRAM
CASE B
EXAMPLE OF RADIOLOGICAL INDICATORS
OF SIGNIFICANT CORE DAMAGE*
CONTAINMENT MONITOR READING ≥2000 R/hr**
DOSE RATE OUTSIDE OF CONTAINMENT HATCH ≥10 R/hr**
OR
GENERAL EMERGENCY
*Presumed to be corroborated (i.e., not merely a faulty reading).
**Site-specific parameters — should be prec. lculated.

Fig. 1.7. Examples of emergency action levels for a General Emergency.

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Slide 40

As an example, Table 1.8 shows typical derived relationships between containment monitor levels and postulated core damage states for a number of light water reactors as reported to the NRC. These would be key indicators of the level of core damage for accident sequences that release into the containment. These plant-specific estimates of containment radiation levels associated with various levels of core damage vary greatly. This variation may be due to (1) basic accident scenario (release) assumptions used for the estimates or (2) assumed monitor efficiency, shielding, and location in the containment. Nevertheless, the containment radiation monitor readings would increase by several orders of magnitude (1000 times) for progressively more severe core damage levels, such that the plant operator could recognize the difference between a minor problem and a major one. Core damage sufficient to declare a General Emergency would involve the release of about 20% or more of the gap inventory. In the examples in Table 1.8, the containment monitors would read between 100 and 20,000 R/hr for this level of damage. For purposes of comparison, the radiation reading for a major leak of primary water into the containment, without significant core is expected 1 to 10 R/hr or less.

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Table 1.8. Typical calculated containment radiation estimates (R/hr)[a]

Plant	Assumptions				Type of reactor
	100% gap activity	1% core activity	10% core activity	100% core activity	
A	400	4,000	40,000	-	Pressurized water
B	1,900	1,900	18,000	200,000	Boiling water
C	165,000	-	-	-	Pressurized water
D	-	5,000	50,000	500,000	Boiling water
E	80,000	150	70,000	70,000	Pressurized water
F	100,000	10,000	100,000	1,000,000	Boiling water

[a]R = roentgen, the international unit of x-radiation or gamma radiation.

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1.4.2 Protective Action Initiation Based on Dose Projections

Slide 41

In the past, considerable attention has been given to the use of dose projections as the only basis for initiating off-site protective actions. However, it should be recognized that, for some very severe accidents, real-time dose projections would be available too late for initiation of an early (most effective) protective response off site. Studies have shown that, for a very severe accident with a major release shortly after core damage, evacuation of the population near the plant (2 to 3 miles) must start before or soon after the release to prevent early severe health effects. (The effectiveness of protective action is discussed in Vol. 4 of this manual.) Early evacuation of all nearby areas would considerably reduce all radiation exposure risks.

Slides 42 and 43

Predicting dose requires several steps: (1) predicting the quantity and timing of the release from the plant (source term), (2) predicting the movement of the plume (transport), and (3) predicting the dose from the plume.

1.4.2.1 Predicting the Release from the Plant

1.4.2.1.1 Predicting the Source Term Following a Severe (Core Melt) Accident

Slides 44 and 45

During a severe (core melt) accident, predicting the release (i.e., the source term) would be the first step in projecting doses. Therefore, any errors associated with predicting the source term would have a direct impact on the

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accuracy of dose projections.

As part of the upgrades that followed the Three Mile Island--Unit 2 accident, on-line radiation monitors capable of measuring the noble gases released through plant vents were installed. On-line monitors for iodine and other particulates were not considered practical. Therefore, the presence of iodine and particulates in a release is determined through analysis of samples taken during the release. This could require several hours. Note that noble gases are not considered as great a threat to the public as radioactive iodine and other particulates. Although current systems can characterize most releases, they cannot provide fast estimates of those very unlikely releases that pose the greatest threat to the public. Even more importantly, the plants are designed to accommodate routine releases of radioactivity and to minimize releases resulting from abnormal conditions and accidents. However, because an accident resulting in off-site early health effects (death and injuries) would have to be fast, direct, and unfiltered, such a release would most probably be via an unmonitored pathway to the atmosphere. The most important example is a release due to a major containment failure or major containment penetration failure. As a result, effluent-monitoring systems located in routinely monitored release pathways (e.g., stacks) will not be able to assess the extent and the characteristics of such a severe release.

For accidents where the total release is through a monitored pathway (e.g., the stack), it may be possible to obtain a good characterization of the release. At a minimum, the magnitude in relative terms (e.g., this release has the possibility to exceed Environmental Protection Agency protective action guides) can be estimated---if the monitors stay on scale. By their very nature, however, releases resulting in off-site dose high enough to cause early health effects most likely cannot be characterized by effluent monitors.

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1.4.2.1.2 Current Reassessment of Potential Source Terms

Slide 46

Since the Three Mile Island—Unit 2 accident, considerable research has been conducted by the nuclear industry and NRC and its contractors on the reassessment of potential source terms. Reassessment of the Technical Bases for Estimating Source Terms (NUREG 0956) describes this effort and states the following interesting conclusions.

- Containment performance (survival, failure, or bypass) is a major factor affecting source term.

- The analyses performed to date suggest that generalizations are inappropriate since large factors of reduction in source term were not found for all sequences (accidents) as reported in other studies.

The study further showed that, even if all plant conditions were known, the current computer models could predict the source term only to within a factor of 100. This highlights the difficulty of source term estimation because, during an actual accident, detailed plant conditions would not be known.

Slide 47

The result is that, early in the response to a severe accident that results in a major release, it will be very difficult to predict the "source term" with a reasonable degree of accuracy.

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Slide 48

Some simple tools have been developed that could at least provide a bound of possible off-site doses, based on consideration of plant (core and containment) conditions. Figure 1.6 is a very simple example. Other examples are the 15 precalculated off-site dose projections for severe accident conditions and for many meteorological conditions presented in Dose Calculations for Severe LWR Accidents (NUREG-1062).*

Such precalculated dose projections must be used with considerable caution, just as with a real-time dose projection, principally because of the major uncertainties of what the source term would be during a severe accident sequence. However, projections of source terms based on consideration of plant conditions have the advantage that all possible accidents and release pathways can be considered.

1.4.2.2 Transport and Dose

Once an estimate has been made of the radioactive material being released from the plant, movement of the material through the atmosphere must be characterized.

Slide 49

Even if one could accurately predict the radioactive material that may be or is being released from a plant during a severe accident, significant uncertainties would still be associated with dose projections to off-site areas. Dose models give a very simple picture of a very complex situation.

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* See Vol. 1, Appendix A, "Bibliography for Volumes 1-5."

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Slide 50

In a 1981 study conducted at the Idaho National Engineering Laboratory, a nonradioactive tracer (SF_6) was released and the resulting air concentrations were compared with predictions made by various models to evaluate their potential use in emergency response situations. Figure 1.8 shows the actual air concentration (plume) pattern observed for one of the tests and the plume pattern predicted by three of the models tested under this program: (a) a simple, straight-line Gaussian plume model of the type used by many emergency response organizations, (b) a Gaussian-puff trajectory model like that used in the NRC emergency response center, and (c) a more sophisticated windfield and topographic model used in the Department of Energy's (DOE's) Atmospheric Release Advisory Capability (ARAC) program. Even the most complicated model could not reproduce what actually occurred.

Slides 51 and 52

This result points out two concerns. First, typically, only one local meteorological tower is in the site vicinity. The initial transport of radioactive material from a site after it is released to the atmosphere will be dominated by local conditions (e.g., hills, valleys, lakes, and precipitation). This single source of weather and wind information cannot give a definitive indication of winds away from the plant.

Nuclear power plants are typically located in very complex areas (e.g., in river valleys or on the coast) where wind direction and flows can vary considerably within a short distance of the plant. As an example, a 180° difference in wind direction could result from sea breeze effects at a coastal site. This is the basis for taking protective actions in all directions near the plant (within 2 or 3 miles). The events that occurred early in the Three Mile Island incident illustrate

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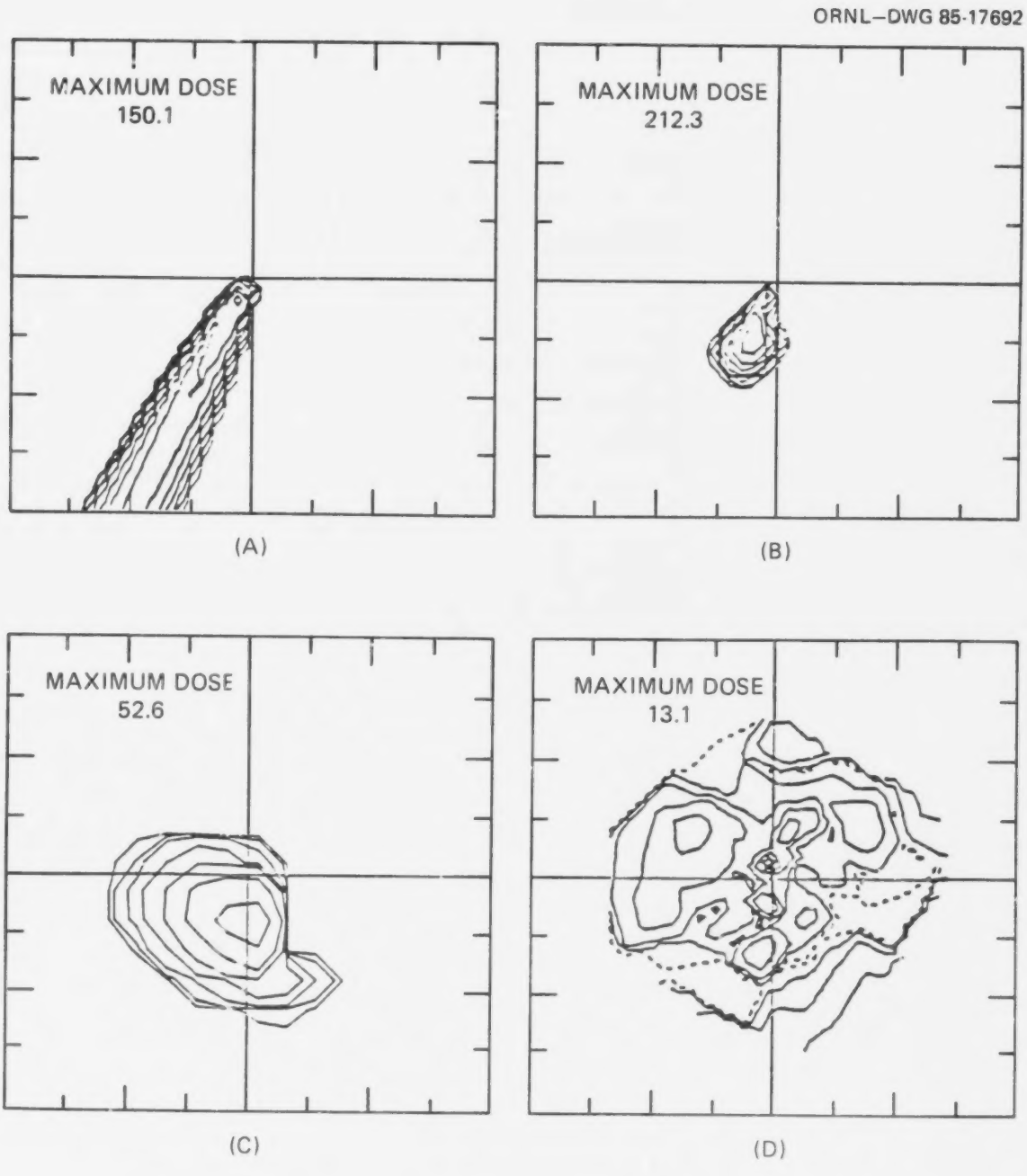

Fig. 1.8. One-hour surface doses predicted by (a) Gaussian plume model, (b) puff-trajectory model, (c) complex numerical model, and (d) doses actually observed.

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the problems inherent in taking protective actions only in the downwind direction. This will be discussed later.

Slide 53

Second, differences should be expected in the estimates produced by various dose models even if the same input conditions (e.g., source terms and meteorology) are used to estimate off-site doses. Various response organizations will also be performing analyses based on different assumptions. For example, the NRC may be concentrating on dose projections based on possible additional plant failures, while the state is making dose projections based on estimates of actual releases. Therefore, a 10- to 100-fold (or greater) spread in calculated doses must be anticipated among the various response organizations (licensee, NRC, state and local officials, and U.S. Department of Energy).

What may be more important than relying on a dose model in estimating plume movement is a knowledge of local meteorological conditions and trends (e.g., the winds shift every morning at about 9:00 a.m.).

The basic point here is that the responder needs to understand the problem, the technical aids, and the results. Indiscriminant use of technical aids such as dose projection models without access to staff who understand the unpredictability of local conditions can provide misleading input to protective action decision making.

As emphasized earlier, in the early emergency response phase, dose projections for protective action decision making should be secondary to assessment of plant conditions and general weather at the time. After a release, environmental monitoring information should form the basis for additional protective actions.

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Slide 54

Environmental monitoring would be the best way to characterize a release after it occurs. However, one must be sensitive to the differences between actual plume behavior and that simulated by models. Dose projection models project the average dose as the result of plume meander over a 15- to 30-min period. Therefore, as can be expected, a monitoring team within the actual plume (Fig. 1.9, point B) may observe greater doses than projected or, if the team is out of the plume (point A), lower doses than projected. Even if the model projections are "correct," actual field-monitoring results could differ considerably from projections because the projections show averages. One would not expect the first preliminary field monitoring results to agree with model projections, even under the best circumstances.

1.4.2.3 Estimates of Overall Uncertainties in Severe Accident Off-Site Dose Projections

Slide 55

Based on conclusions drawn from the preceding discussions about the difficulties of estimating source term and plume movement, an overall estimate of the uncertainties associated with dose assessment for severe accidents has been made by the NRC technical staff. These estimates, given in Table 1.9, are estimates of the ratio of what a model may project for an accident sequence and what the actual average dose rate may be. It is apparent that, overall, the best that should be expected in the early time frame is that projected dose estimates may be within a factor of 10 of the true dose value; more likely, they will be even less accurate. Table 1.9 also shows that the largest single component of uncertainty is expected to be the uncertainty in the estimate of the source term. Unanticipated catastrophic containment failure is an example of a case where

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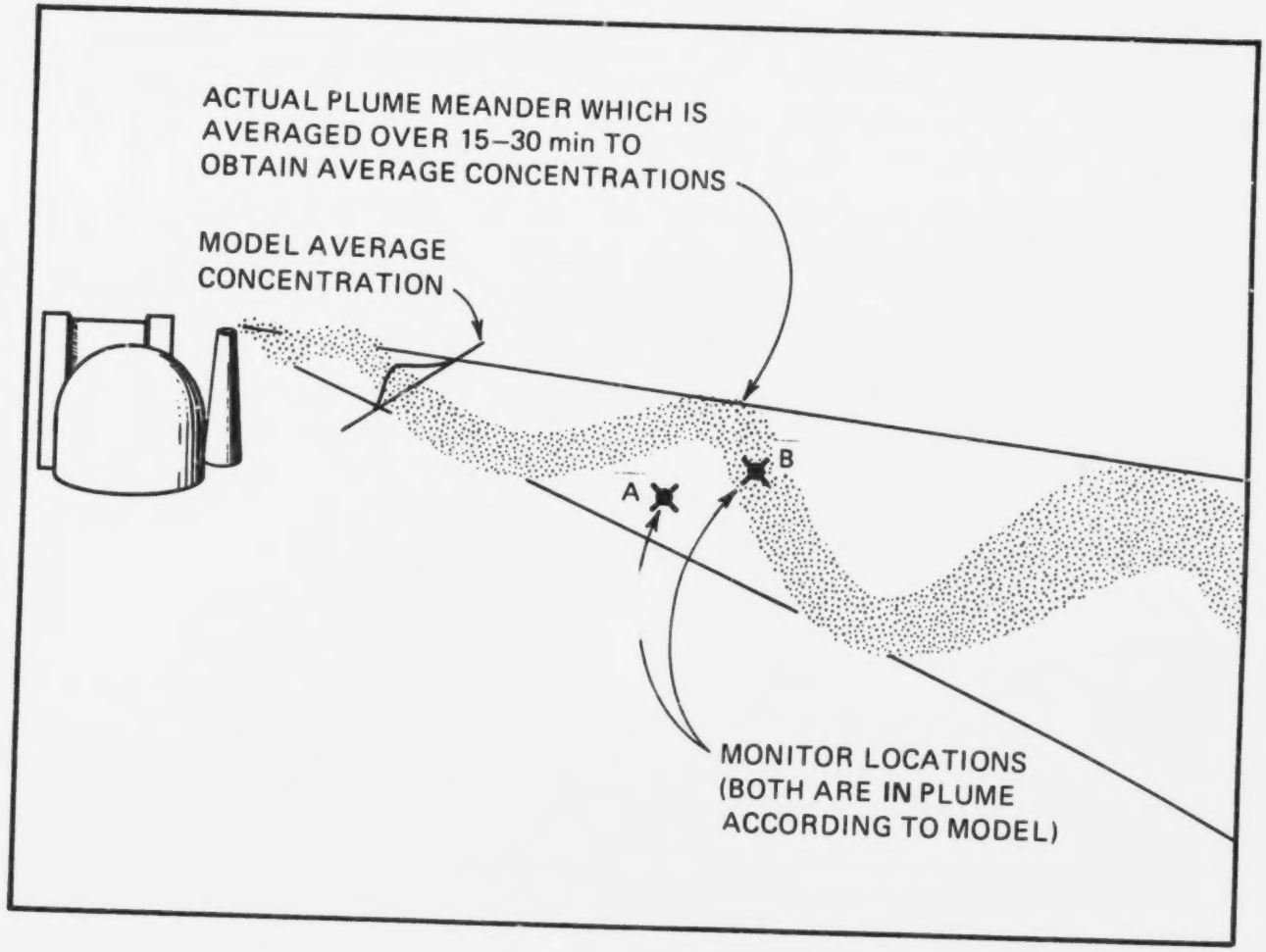

Fig. 1.9. Relationship between an actual plume and model projections.

Table 1.9. Estimated range of uncertainty between early projected dose and actual off-site dose for a severe accident (core melt)[a]

Element	Uncertainty factor[b]		
	At best	Most likely	Near worst
Source term (event and sequence)	5	100–1000	100,000
Dispersion			
Diffusion (concentration)	2	5	10
Transport (direction)	22°	45°	180°
Transport (rate)	1	2	10 (low wind speed)
Dosimetry	3	4	5
Overall (dose and direction)	10, 22°	100–10,000, 45°	100,000, 180°

[a]These estimates are for an averaged dose at a location (e.g., over 15 to 30 min), not for a specific or single monitor reading.

[b]Ratio of a likely maximum or minimum value to the expected median value.

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source term could be underestimated by a factor of 100,000.

For lesser accidents (non-core damage) where the total release is through a monitored pathway and consists mostly of noble gases, the source term uncertainty can be reduced. However, the transport and dose uncertainties would remain unchanged.

It is clear that one should not expect close agreement when comparing various dose projections with each other or with early field monitoring data. Dose projections should be viewed only as rough estimates.

1.4.3 A Bad Example: Early Protective Action Decisions During the Three Mile Island--Unit 2 Accident

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To highlight some of the points discussed in this section, certain aspects of the assessments of the Three Mile Island--Unit 2 accident merit discussion. Figure 1.10 presents the hourly wind vector as measured by the site meteorological system during the first day of the accident.* It is evident that wind direction at the site varied dramatically throughout the 12-hr period. Between 7:30 and 8:00 a.m., the State of Pennsylvania issued warnings of imminent evacuation to the west of the site. (A Site Emergency was declared at 7:20 a.m., followed by a General Emergency at 7:30 a.m.) At 8:10 a.m., this preparedness was reduced to a standby notice because dose rate measurements to the west were "only" about 30 millirem/hr (i.e., about 10,000 times higher than the dose rate resulting from normal effluent releases). This reduction-to-standby notice came coincident with core uncovering.

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* Actually, these measurements were not available to the NRC until three days later because the plant computer crashed early in the accident progression.

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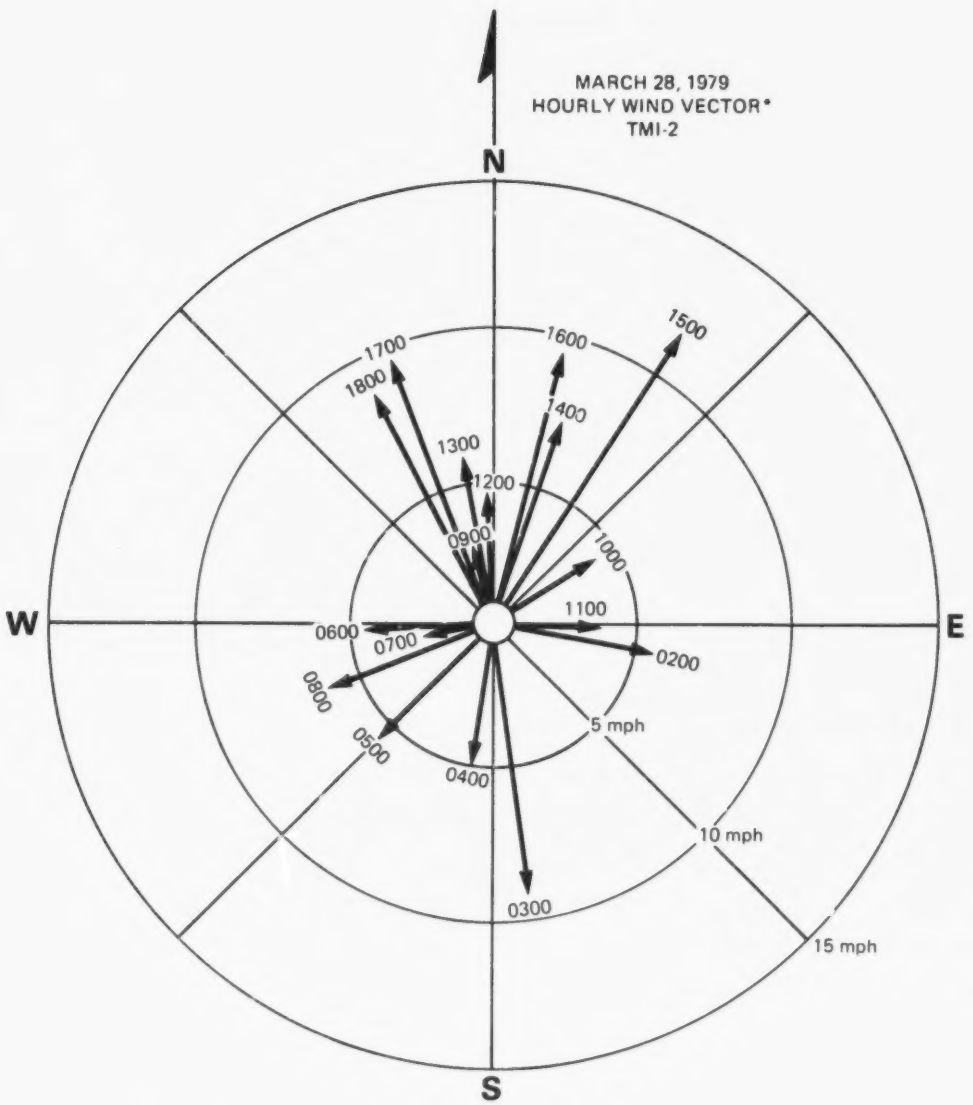

*Arrows indicate direction toward which the on-site wind was blowing at the local time indicated. Circles represent varying wind speeds.

Fig. 1.10. Hourly wind vector at Three Mile Island on March 28, 1979.

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Moreover, even if an evacuation to the west of the site had been initiated around 8:00 a.m., local wind conditions would have shifted the potentially affected area to the north by 9:00 a.m., and then to the east by 11:00 a.m. Thus, the wrong people would have been told to evacuate.

As the NRC Special Inquiry Group noted later, based on in-plant observations as set forth in the emergency plans and as emphasized in NRC emergency planning guidance in place even at the time (R.G. 1.101), omnidirectional evacuation of the total low-population zone (2.5-mile-radius area surrounding the site) would have been warranted no later than 7:30 a.m.

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By 9:00 a.m., indications of severe core damage were indisputable. Some of the core thermocouples showed temperatures over 2000°F (800°F beyond that required for cladding failures--see Fig. 1.7), and the containment dome monitor increased from 600 to 6000 R/hr between 8:20 and 9:00 a.m. However, as indicated, the decision not to take action was made based on field-monitoring results. The NRC Special Inquiry Group found that the state offices should have been advised at 9:00 a.m. that "the core has been badly damaged and has released a substantial amount of radioactivity. The plant is in a condition not previously analyzed for cooling system performance."

The Inquiry Group went on to state, "The difficult question in this situation is whether to advise precautionary evacuation of the nearby population or to advise only an alert for possible evacuation. The recommendation to evacuate is consistent with what we think would then be the case, a prudent doubt that the core-cooling passages were still sufficient for cooldown. In addition, the containment building was now filling with intensely radioactive gas and vapors, leaving the nearby public protected

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by only one remaining barrier, the containment, a barrier with a known leak rate that needed only internal pressure to drive the leakage."

Finally, the Inquiry Group stated, "Present emergency plans are inadequate because they do not provide a clear requirement to evaluate the need for protective actions based on deterioration of plant conditions."

This example illustrates the importance (for core melt accidents) of implementing protective actions in the nearby areas as soon as core damage is detected and without regard for wind direction or detection of actual major releases. These are two of the foundations of current NRC staff emergency planning guidance. Early precautionary evacuation of the immediate area (approximately 2-mile radius) should not be recommended in only "downwind" directions because of the inability to determine where downwind will be when the protective actions are actually implemented or when a significant release occurs. In addition, when core damage is detected, the early recommendation to evacuate should not be based on early real-time dose projections but on the status of the core. Indeed, the predetermined, early, initial evacuation for a General Emergency is called "precautionary" because a major release may never actually occur, as was the case at Three Mile Island--Unit 2. On the other hand, no immediate, early evacuation would be warranted for sequences less serious than core-melt accidents.

1.4.4 Role of Off-Site Dose Projection in Severe Accidents

Slide 58

The role of dose assessment during a nuclear power plant accident will depend on the type and phase of the accident. Dose assessment during the initial phase of a severe accident (General Emergency) provides a basis to establish priorities for the use of limited resources in the implementation of off-site actions such as deployment of field-monitoring teams. In an

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actual uncontrolled release of radioactive material to the environment, it would be imperative to obtain off-site monitoring team data as quickly as possible. However, for a core-melt sequence, early protective actions in nearby areas (2 to 3 miles) should not await such results. In particular, the evacuation of nearby areas for a core-melt accident sequence (i.e., General Emergency) should be initiated on the basis of plant conditions.

After implementation of protective actions near the plant (based on an assessment of plant conditions), dose projections should be used to determine whether these actions should be extended. The model projections may indicate the maximum distance from the plant where further actions are required. However, because of the difficulty of projecting plume movement, actions should not be limited to just the downwind areas. Actions should be taken in all directions or at least in all areas considered to be the possible limits of the plume under various conditions (e.g., inside a valley).

Bounding dose calculations may be very useful in comparing the consequences of various plant response options (e.g., venting the containment vs allowing later containment failure).

For Site Area emergencies and for the intermediate phase of a General Emergency (after early protective actions have been implemented for the population near the plant), the roles of dose assessment differ significantly from those discussed earlier. The first role is to assess the areas that may warrant implementation of protective actions according to radiation protection objectives. The second role of dose assessment under these conditions is to provide feedback regarding the magnitude and composition of a release based on the analysis of off-site

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samples and monitoring results.

1.5 ROLE OF FIELD MONITORING

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Actual field monitoring can be used to determine accurately the actual dose rates and projected doses off site as the result of an accident. As discussed in Vol. 4, the role of field monitoring during the early phases of a severe accident would be to identify areas that may require further protective actions following a release. Reliance should be on field monitoring as soon as possible.

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Because the actual location of the plume or resulting ground contamination may not be known for some time, early, limited field-monitoring results should be used with great care. Even if the monitoring team is in the path of the release, the plume could be overhead (as shown in Fig. 1.9) or could meander or loop around the team. It will be difficult to obtain readings that are considered representative of the release.

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The biggest problem with field monitoring, as shown in Fig. 1.8d, is that the actual distribution of off-site dose could be very complex. The dose rate could change over very short distance (hundreds of feet). "Hot spots" could be surrounded by areas of lower dose rates. Therefore, an aircraft or large numbers of monitor teams would be needed to fully

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characterize a major release in a short time.

The teams should have instruments designed to monitor all radioactive material (iodine, cesium, strontium, and tellurium) that may be released during an accident. If air samples are taken, this analysis could take several hours. Monitoring teams typically will be dispatched into the emergency planning zone (EPZ) within an hour after initiation of a severe release. This subject is elaborated upon in Vol. 4 of this manual.

1.6 MAJOR POINTS

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The major points covered in this section are summarized as follows:

- Accidents or incidents less grave than significant core damage, or the imminent threat thereof, would not warrant predetermined protective actions off site.

- Keeping a core cool after scram is fundamentally a simple chore; it involves using decay heat to boil water at a rate of 1000 gal/min or less [1000-MW(e) plant].

- The release to the atmosphere of even a small fraction of the inventory of radionuclides in the core of any operating reactor would result in significantly elevated risks of adverse health effects off site.

- Core damage and a fast, direct release pathway are required

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for induction of early fatality and injury off site.

- There would be clear indicators in the control room of actual core damage.

- Early off-site protective actions must be driven by knowledge of in-plant conditions, especially conditions of the core. These initial actions are taken based on predetermined in-plant emergency action levels. Basically, core damage should warrant that people near the plant evacuate immediately.

- It is not wise to await a major release to the atmosphere (i.e., a major containment failure) before making protective action recommendations to the public.

- The control room staff will be very busy during a severe accident.

- Early in a severe core-melt accident, it would be difficult, if not impossible, to make a confident projection of off-site doses.

- Protective actions near the site (2 to 3 miles), if warranted at all, should be implemented in all directions--not just in the downwind direction.

- Dose projections and actual field measurements will differ considerably, even if the dose projection model is doing a good job.

- Results of various dose models may be considerably different, even if each model is using the same inputs.

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- For the initial stages of a severe core-melt accident (General Emergency), off-site dose projection has a secondary role, independent of initiating protective actions near the plant.

- Field-monitoring results would be the most accurate indicator of off-site radiological impacts and their extent, but early field-monitoring results should be used with caution.

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Appendix A

SLIDES RELATING TO VOLUME 2 OF THE SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL

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APPENDIX A

SLIDES RELATING TO SEVERE REACTOR ACCIDENT INCIDENT RESPONSE TRAINING MANUAL:

SEVERE REACTOR ACCIDENT OVERVIEW

VOL. 2

NUREG-1210

ORNL/TM-9271/V2

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OBJECTIVES

- Describe the dose levels required to produce early health effects
- Describe basic radiation protection criteria, the EPA and FDA PAG levels, their relationship to health effects, and how they are used
- Describe the amount of radioactive material that must be released to produce early health effects and to exceed the EPA PAGs at a distance of about 1 mile from the plant
- Describe the location in the plant of the radioactive material required for early health effects, the damage required for its release, the means for detection of this damage, and the level of certainty of the detection
- Describe conditions in the control room during the initial phase of an accident, including fundamental tasks the control room staff must accomplish

Slide 2

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OBJECTIVES (continued)

- Describe how a knowledge of plant conditions (e.g., core temperatures) is essential in projecting consequences of an accident and why in-plant monitors cannot be relied upon entirely to estimate plant releases
- Describe the uncertainties associated with dose assessment, including those between (A) what is predicted and what may be actually measured and (B) different model estimates
- Describe why it may be impossible to adequately project dose during the early phase of a severe accident
- Describe the role of dose projection during a severe nuclear power plant accident (i.e., general emergency) and during lesser events
- Describe why early protective actions for the population at greatest risk (those near the plant) should always be recommended in all directions near the plant and not just downwind

Slide 2 (Continued)

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LIGHT WATER REACTORS (LWRs)

- Pressurized water reactors (PWR) (70+)
- Boiling water reactors (BWR) (20+)
- Major release from fuel possible in minutes following the loss of systems designed to keep the core cool

HIGH-TEMPERATURE GAS-COOLED REACTOR (HTGR) (only 1)

- Hours before a major release would be possible following the loss of systems needed to keep the core cool

BASIC SAFETY PROBLEMS

Reactor operation creates

- Radioactive fission products that must be contained
- Decay heat that must be removed or else it could drive the fission products into the environment

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CURIE

- Measure of radioactivity
- Not directly related to biological damage – dose

REM

- Measure of potential biological damage
- Unit of dose

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SUMMARY OF VOLATILITY OF RADIOACTIVE MATERIAL IN LARGE [3300-MW(T)] LWR CORE

Volatility	Inventory (in millions of curies)
Very volatile–100% release from fuel possible	
Xenon (Xe) and krypton (Kr)–noble gases	400
Iodine (I)	750
Cesium (Cs)	15
Moderately volatile–1–20% release from fuel possible	
Tellurium (Te)	200
Strontium (Sr) and barium (Br)	370
Not volatile–small percent release possible	
Other	3500

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TYPICAL INVENTORIES OF NOBLE GASES AND IODINE IN REACTOR SYSTEMS

Locations	Inventories in Curies	
	Noble Gas (Xe, Kr)	Iodine (I)
Reactor core total	400,000,000	750,000,000
Reactive core gap	30,000,000	14,000,000
Spent fuel storage pool	1,000,000	500,000
Primary coolant	10,000	600
PWR–other systems		
Waste gas storage tank	100,000	1
BWR–other systems		
Steam line	10,000	25
Waste gas treatment system	5,000	0.25
Shipping cask	10,000	1

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RADIATION DOSE PATHWAYS

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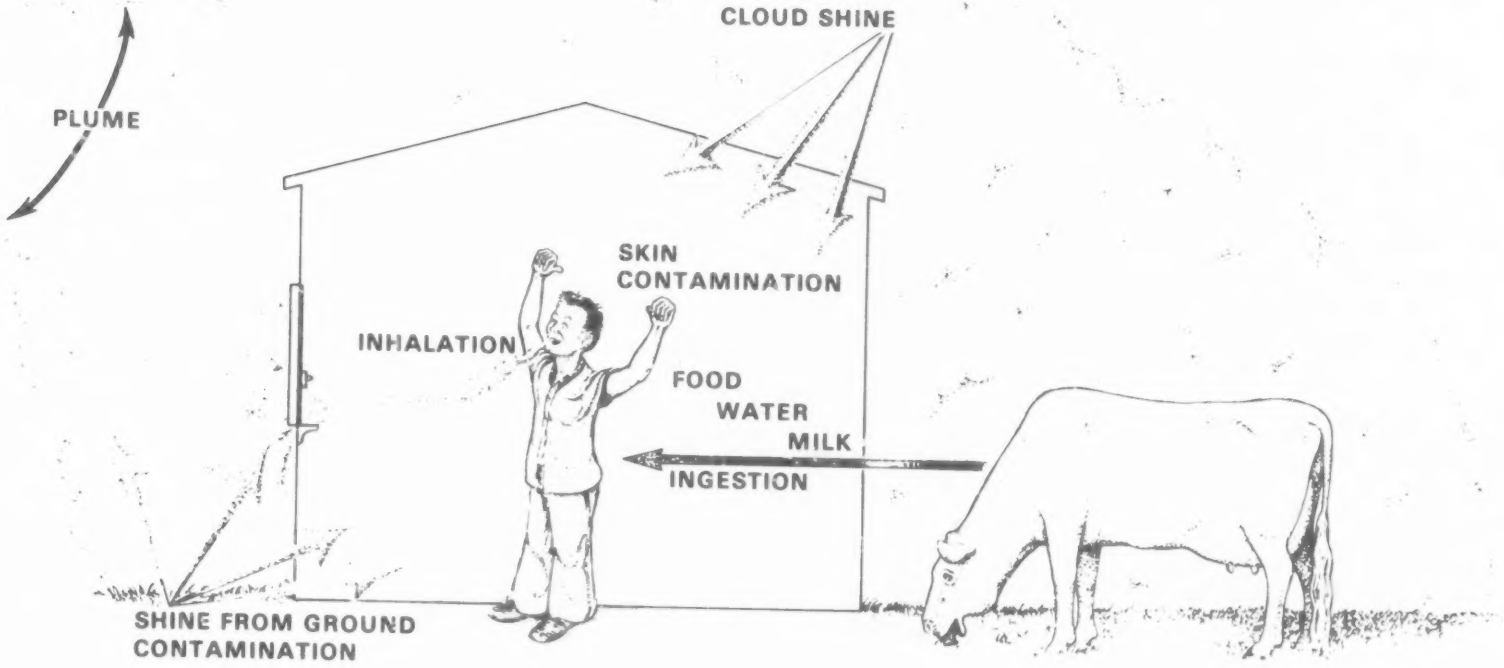

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HEALTH EFFECTS FROM RELEASES OF RADIOACTIVE MATERIAL

Effects	Dose Required
Early injuries	50–100 rem (whole body or bone)
Early deaths	200–600 rem (whole body or bone)
Latent health effects	5000 person-rem[a]

[a]Collective dose (*not* individual dose).

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EPA AND FDA PAGs

Organ	EPA PAGs for Plume Exposure (rem)	FDA PAGs for Food and Agricultural Products (rem)
Whole body (bone)	1–5	0.5–5
Thyroid	5–25	1.5–15
Other body organs	1–5	0.5–5

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ATMOSPHERIC RELEASE (curies)
UNDER POOR METEOROLOGICAL CONDITIONS
TO RESULT IN PAGs AT 1 MILE

Radioactive Material	Pathway	No. of Curies Released	
		15 rem-Thyroid	5 rem-Whole Body
Iodine (^{131}I)	Milk ingestion	2	
	Inhalation	600	
Noble gases (Xe, Kr)	Cloud shine		1,500,000

TYPICAL RELEASES

Sources	Curies Released	
	Noble Gases	Radioiodine
BWR		
Annual total, Ci	100,000	1
Release rate to equal annual total,[a] Ci/hr	10	0.0001
PWR		
Annual total, Ci	200	0.02
Release rate to equal annual total,[a] Ci/hr	0.02	0.000002

[a]Using 10,000 hr/year.

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1-HOUR RELEASE REQUIRED TO EXCEED PAGs

	Release (Ci/hr)
EPA whole-body PAG via cloud shine	1,500,000
FDA thyroid dose via inhalation	600

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SYSTEM FAILURES REQUIRED (BUT NOT NECESSARILY SUFFICIENT) FOR VARIOUS CONSEQUENCES AT 1 MILE

System failure	Is the failure necessary to		
	Cause early health effects?	Exceed whole-body PAGs?	Exceed thyroid PAGs?
Core fuel	Yes	Yes	Yes
Spent fuel	No	Occasionally[a]	Yes
Primary coolant	No	No	Yes
Waste tank	No	No	Yes[b]

[a]The fuel is relatively cool, so the driving force is small.

[b]Milk only.

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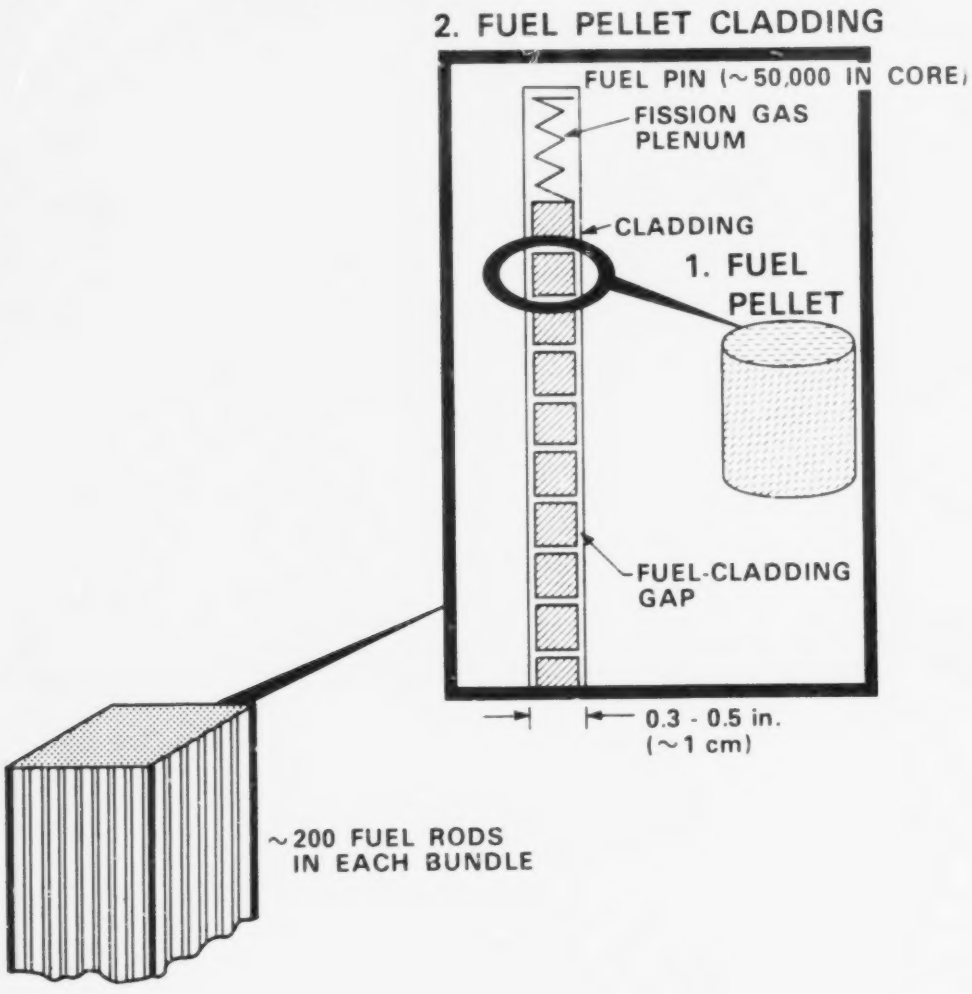
BARRIERS TO RELEASE OF RADIOACTIVE MATERIAL FOR A PWR



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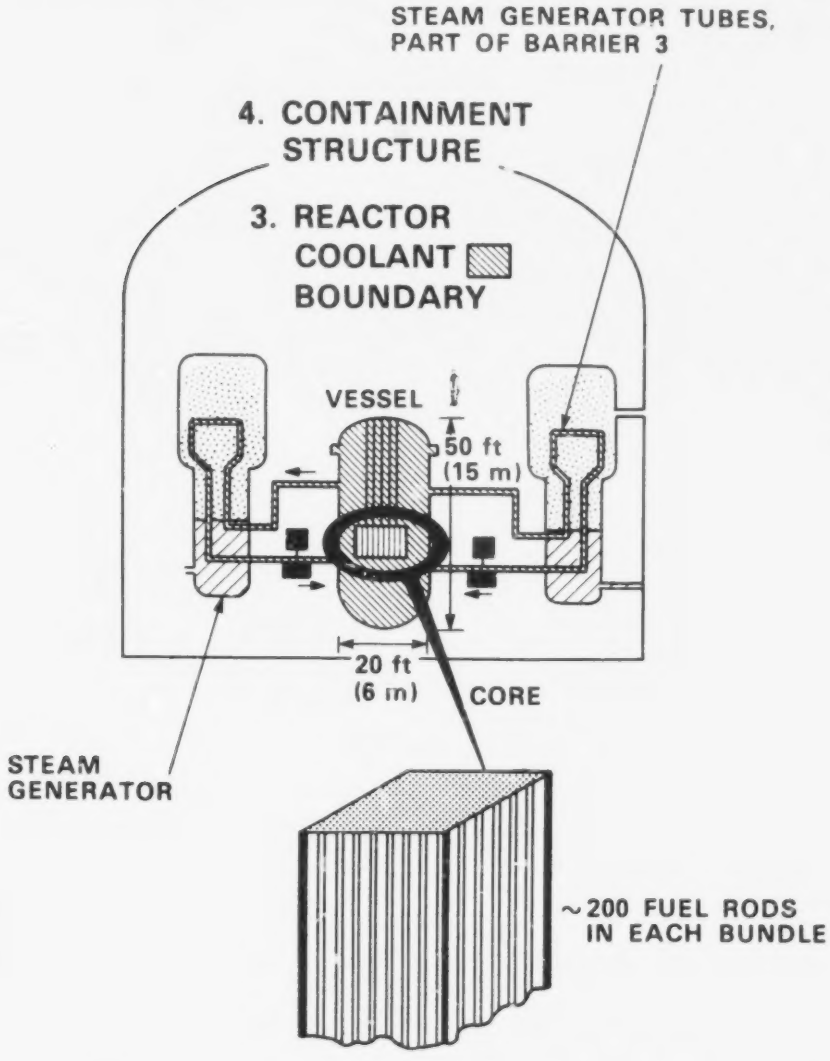
BARRIERS TO RELEASE OF RADIOACTIVE MATERIAL FOR A PWR (continued)



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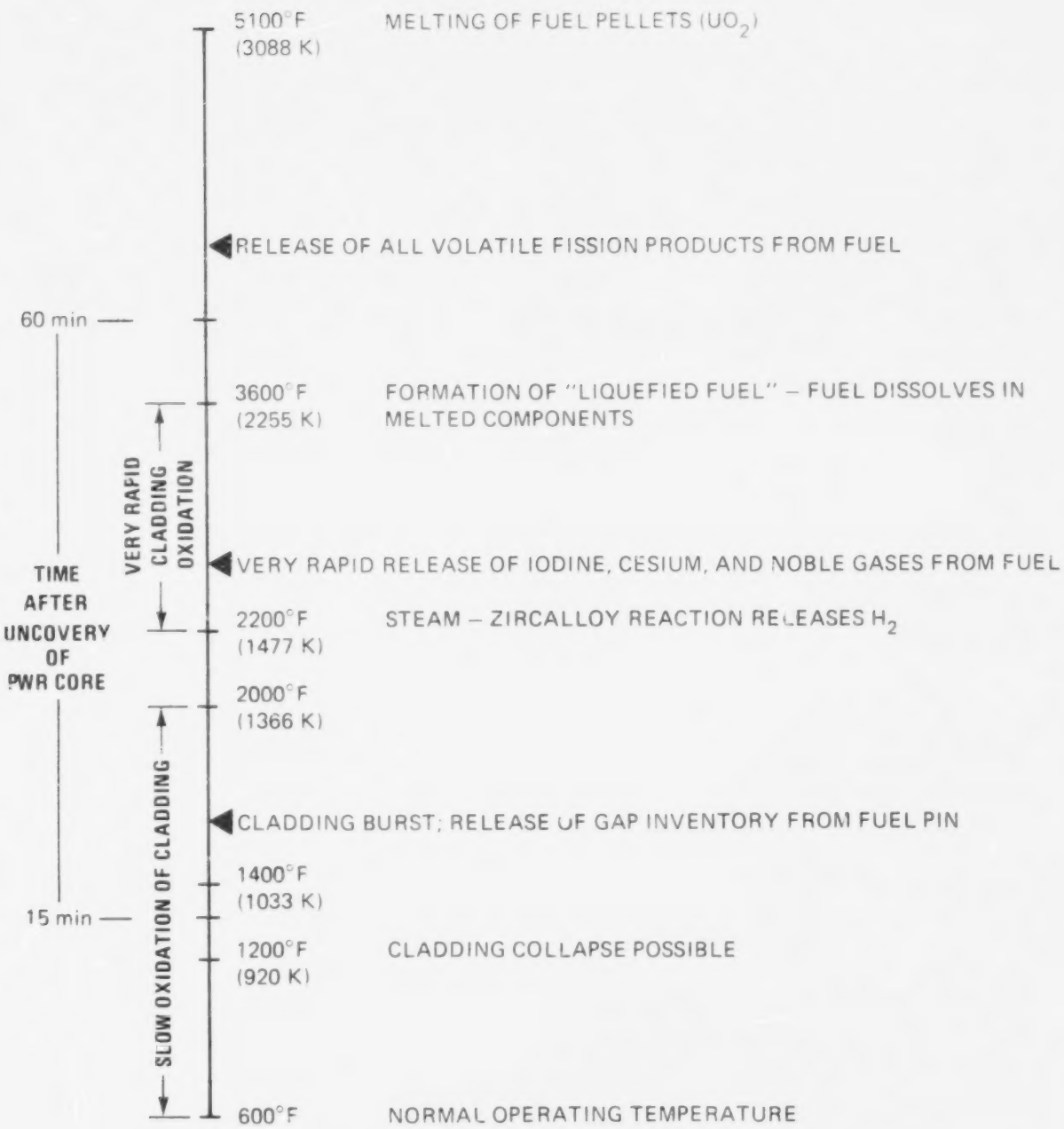
POSSIBLE CORE DAMAGE STATES AS CORE TEMPERATURE INCREASES



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CRITICAL SAFETY FUNCTIONS (EMERGENCY OR NOT)

- Control reactivity
- Maintain coolant level (water)
- Provide coolant makeup (water)
- Remove heat

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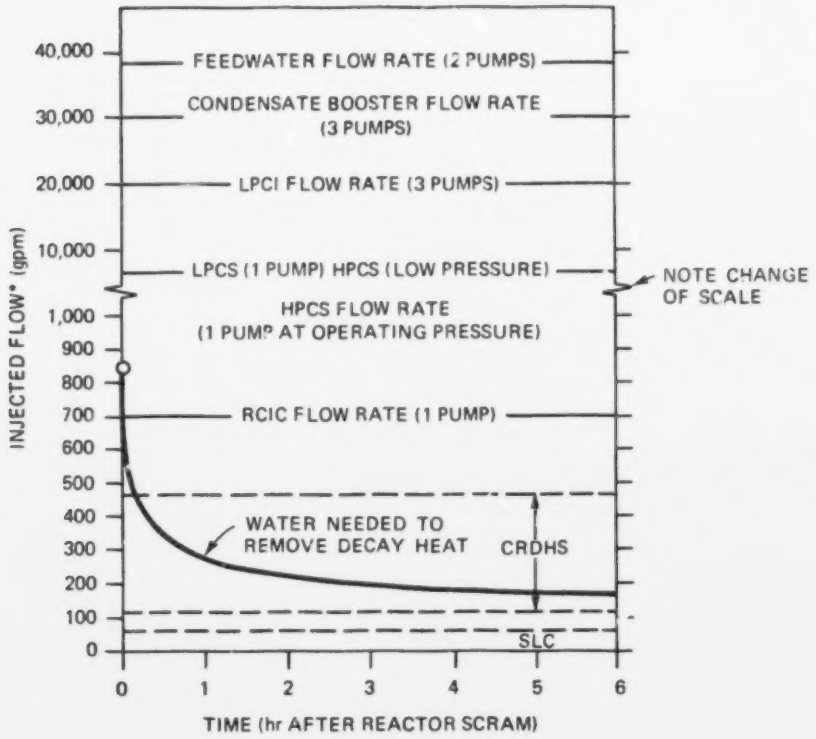
INJECTED REACTOR VESSEL FLOW REQUIRED TO REMOVE DECAY HEAT BY BOILING WATER [3300-MW(T)] AND BWR SYSTEMS AVAILABLE TO PROVIDE THE FLOW



*CALCULATED ASSUMING *CONSTANT REACTOR VESSEL PRESSURE* AND 100% INITIAL POWER DECAY HEAT PER 1979 ANS STD. WITH ACTINIDE DECAY). COOLANT INJECTED AT 305 K (90°F) AND REMOVED AS DRY SATURATED STEAM.

LPCI – LOW-PRESSURE COOLANT INJECTION
RCIC – REACTOR CORE ISOLATION COOLING
LPCS – LOW-PRESSURE CORE SPRAY
HPCS – HIGH-PRESSURE CORE SPRAY
CRDHS – CONTROL ROD DRIVE HYDRAULIC SYSTEM
SLC – STANDBY LIQUID CONTROL

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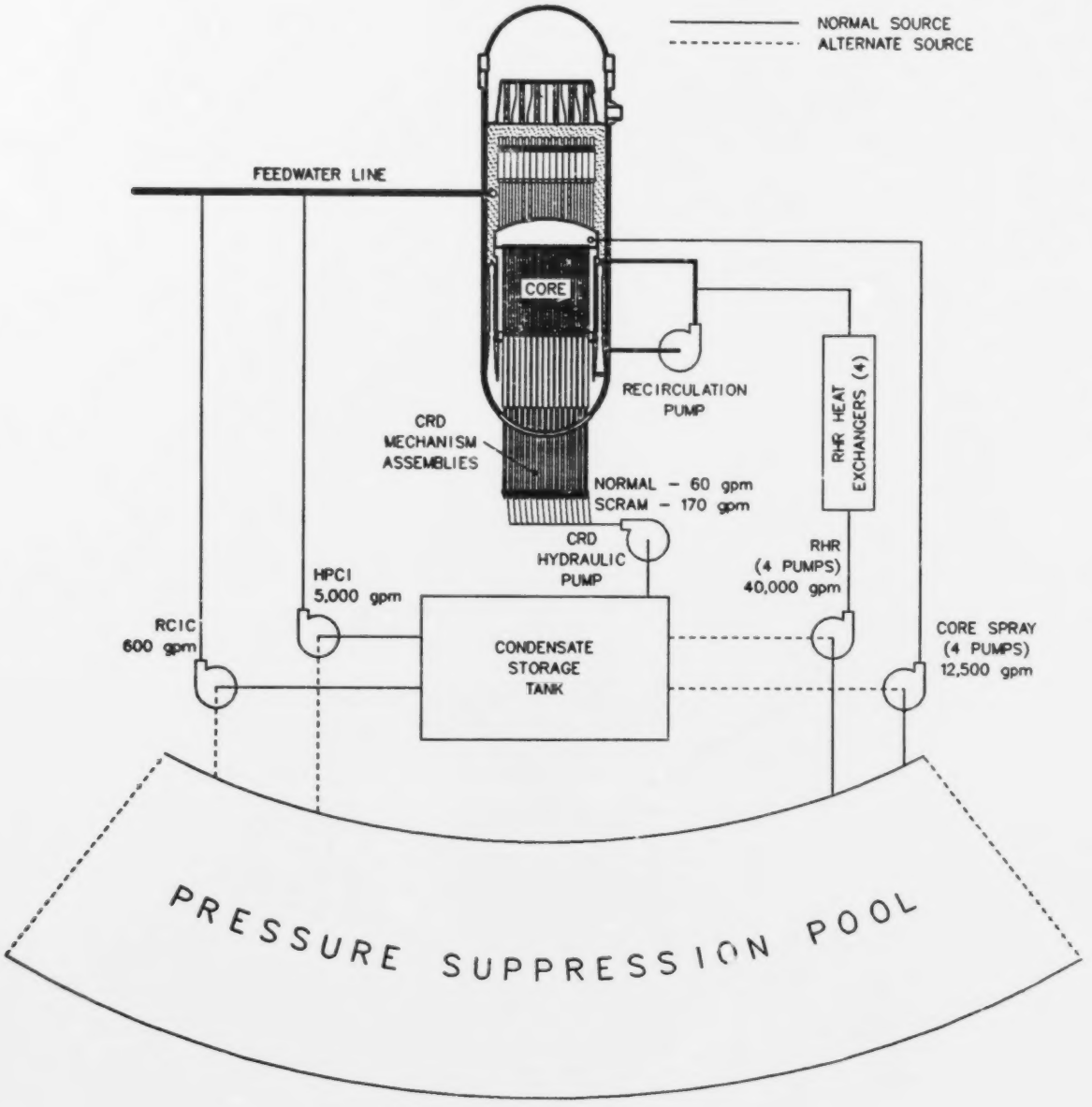
SIMPLIFIED DIAGRAM OF TYPICAL BWR REACTOR VESSEL WATER INJECTION SOURCES



CRD – CONTROL ROD DRIVE
HPCI – HIGH-PRESSURE COOLANT INJECTION
RCIC – REACTOR CORE ISOLATION COOLING
RHR – RESIDUAL HEAT REMOVAL

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EXAMPLES OF INSTRUMENTATION AND INFORMATION AVAILABLE FOR DETERMINATION OF THE STATUS OF THE FUEL (CORE)

Instrumentation and Information	Type of Reactor
Immediate available in control room	
Core temperatures-thermocouples	PWR
Core-exit temperature levels	BWR
Containment radiation level	BWR, PWR
Radiation levels from condenser/air ejector	BWR, PWR
Neutron fluxes in core	BWR, PWR
Available after several hours	
Concentration or radiation level in circulating reactor coolant	BWR, PWR
Analysis of primary coolant-gamma spectrum	BWR, PWR
Containment hydrogen level (from samples)	BWR, PWR

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PLANT CONDITIONS REQUIRED FOR EARLY HEALTH EFFECTS OFF SITE

- Core damage and
- Fast-direct release to atmosphere
  - — Failure of safety systems (e.g., containment sprays)
  - — Early major containment failure
  - — Early major release (less than 4–6 hr)

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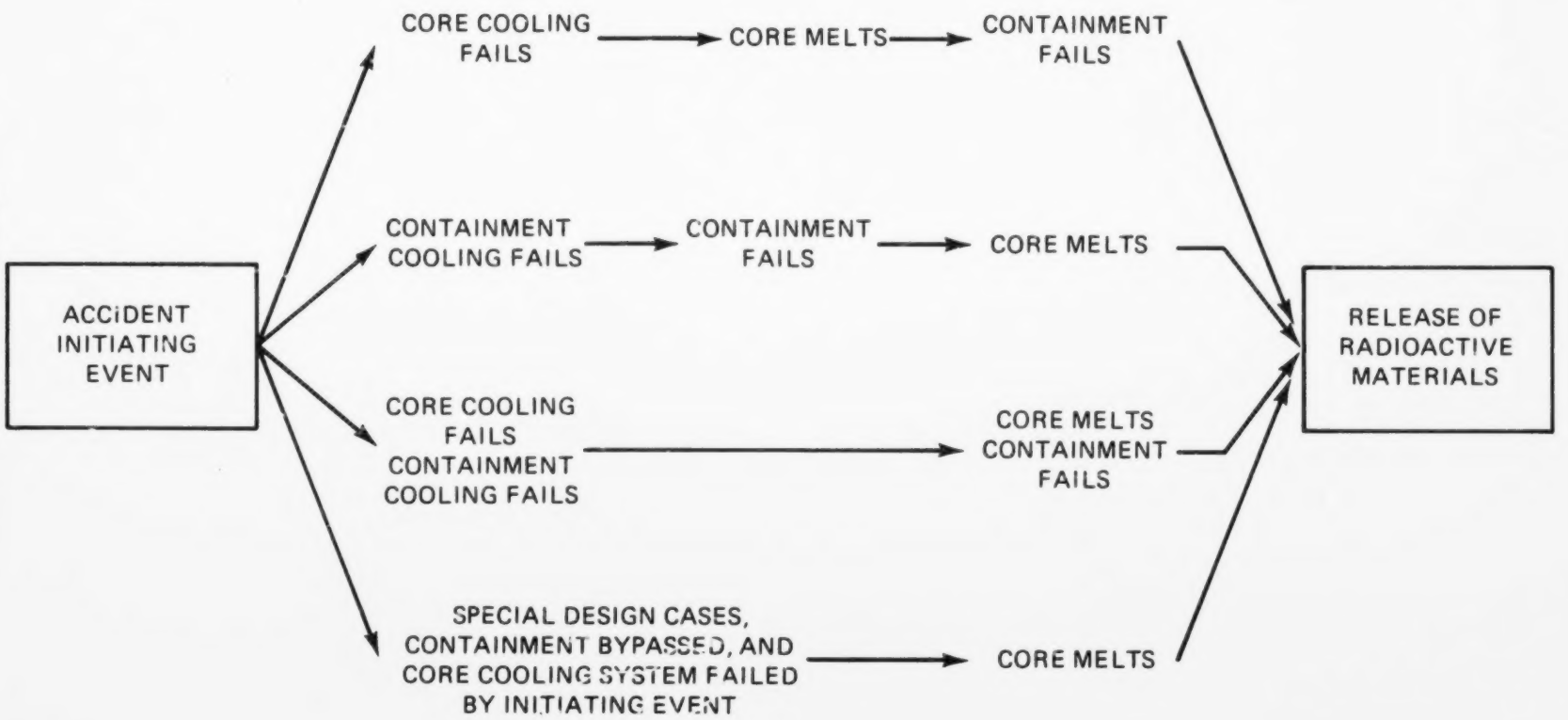
MAJOR SEQUENCES IN A SEVERE LWR ACCIDENT SCENARIO



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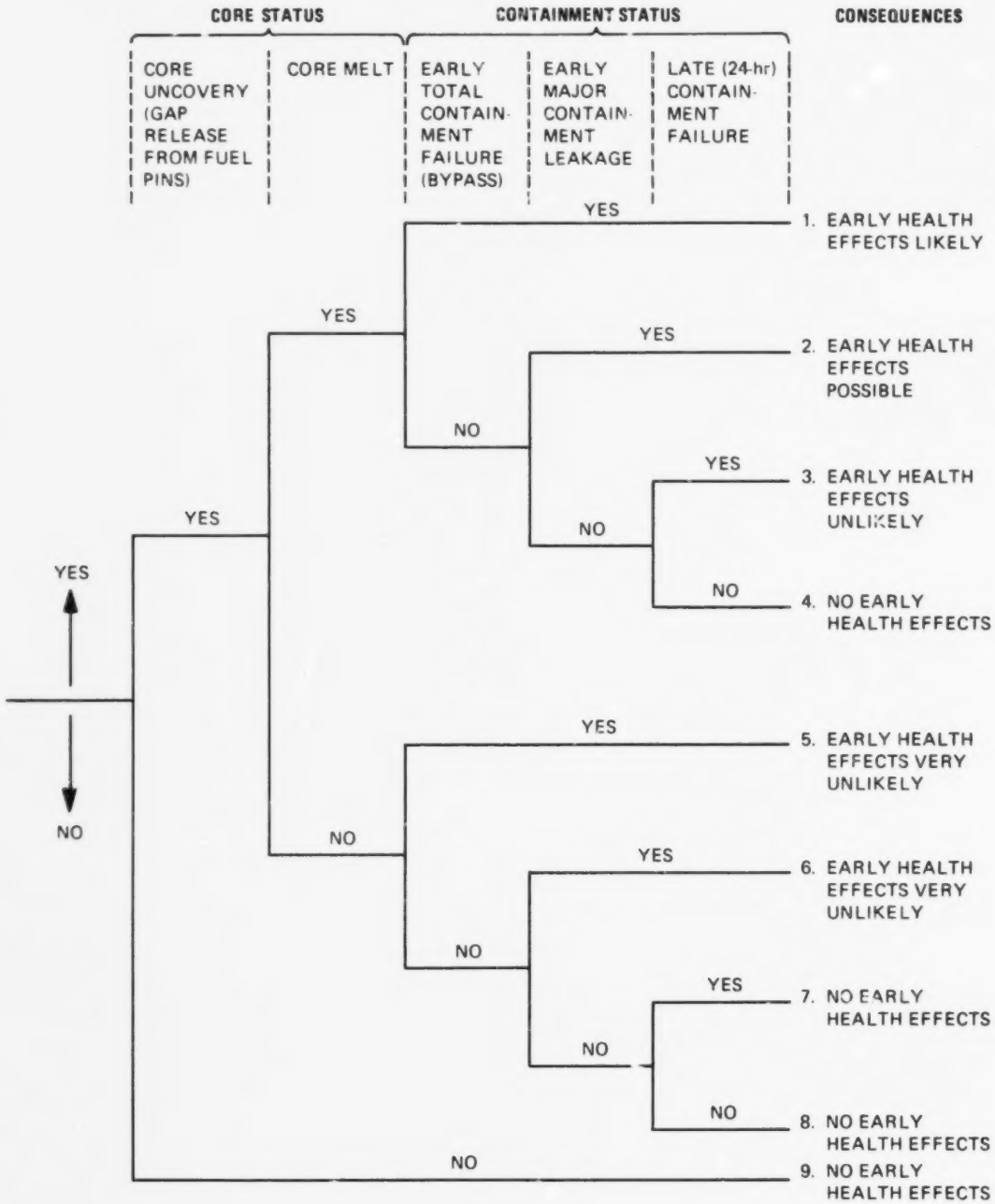
EVENT TREE FOR SEVERE ACCIDENT CONSEQUENCES


CORE STATUS
CONTAINMENT STATUS
CONSEQUENCES
CORE UNCOVERY (GAP RELEASE FROM FUEL PINS)
CORE MELT
EARLY TOTAL CONTAIN-MENT FAILURE (BYPASS)
EARLY MAJOR CONTAIN-MENT LEAKAGE
LATE (24-hr) CONTAIN-MENT FAILURE
YES
NO
1. EARLY HEALTH EFFECTS LIKELY
2. EARLY HEALTH EFFECTS POSSIBLE
3. EARLY HEALTH EFFECTS UNLIKELY
4. NO EARLY HEALTH EFFECTS
5. EARLY HEALTH EFFECTS VERY UNLIKELY
6. EARLY HEALTH EFFECTS VERY UNLIKELY
7. NO EARLY HEALTH EFFECTS
8. NO EARLY HEALTH EFFECTS
9. NO EARLY HEALTH EFFECTS

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CONTAINMENT LOADS WORKING GROUP CONCLUSION

- *Early* containment failure is unlikely during most core melt accident sequences until several hours after core damage

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TYPICAL CONTAINMENT DESIGN PARAMETERS

Type of Plant	Allowable Leak Rate (vol %/day)	Design Pressure (psig)	Estimated Failure Pressure (psig)
PWR, large dry	0.1	47	134
PWR, subatmospheric	0.1	45	119
PWR, ice condenser	0.25	12	50
BWR Mark I	0.5	62	117
BWR Mark II	0.5	55	140
BWR Mark III	0.4	15	60

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SOME IMPORTANT CONTRIBUTORS TO CONTAINMENT FAILURE

- Most important accidents
  - — Overpressurization several hours after core damage
  - — Hydrogen burn
- Some failures before core damage or unpredictable failures
  - — Isolation without scram
  - — Failure to isolate
  - — Containment bypass
  - — Interfacing system loss-of-coolant accident (LOCA)

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REACTOR CONTROL ROOM ENVIRONMENT DURING A SEVERE ACCIDENT

- Plant beyond design
- Many alarms may be sounding simultaneously
- Instruments may respond erroneously or go off scale
- Reactor operators perform many tasks almost simultaneously (first hour)
  - — Identify the basic cause of the problem
  - — Implement emergency operation procedures to control/correct problem
  - — Direct damage control/monitoring teams
  - — Classify the event
  - — Possibly implement on-site emergency action (e.g., evaluation and accountability)
  - — Activate the site emergency organization
  - — Determine and recommend protective actions
  - — Notify the NRC and many other agencies
  - — Man the emergency notification system (ENS) and provide information to NRC

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MOST IMPORTANT ASPECTS OF PREDICTING OFF-SITE CONSEQUENCES

- Little risk if core is not damaged
- Core damage is the result of major equipment or human failures
- No assurance that further failures are not possible
- No assurance that release will be prevented
- Estimate 1 in 10 core damage accidents will result in major releases

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BASIS FOR INITIAL (EARLY) SEVERE ACCIDENT PROTECTIVE ACTION DECISIONS

- Plant conditions (general emergency action levels)
- Precalculated dose projections
- *Not* real time dose projections

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WHY PROTECTIVE ACTIONS ARE BASED ON CORE CONDITIONS

- High risk of major release for core damage accidents
- Ease of detection of core damage
- Difficulty in projecting containment failure
- Effective protective actions are taken before containment failure

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THE PROBLEM

- Protective actions for severe accidents (core damage) must be initiated before a major release to be effective
- Cannot miss core damage from control room
- Cannot confidently dismiss or predict major release (plant is well beyond design)

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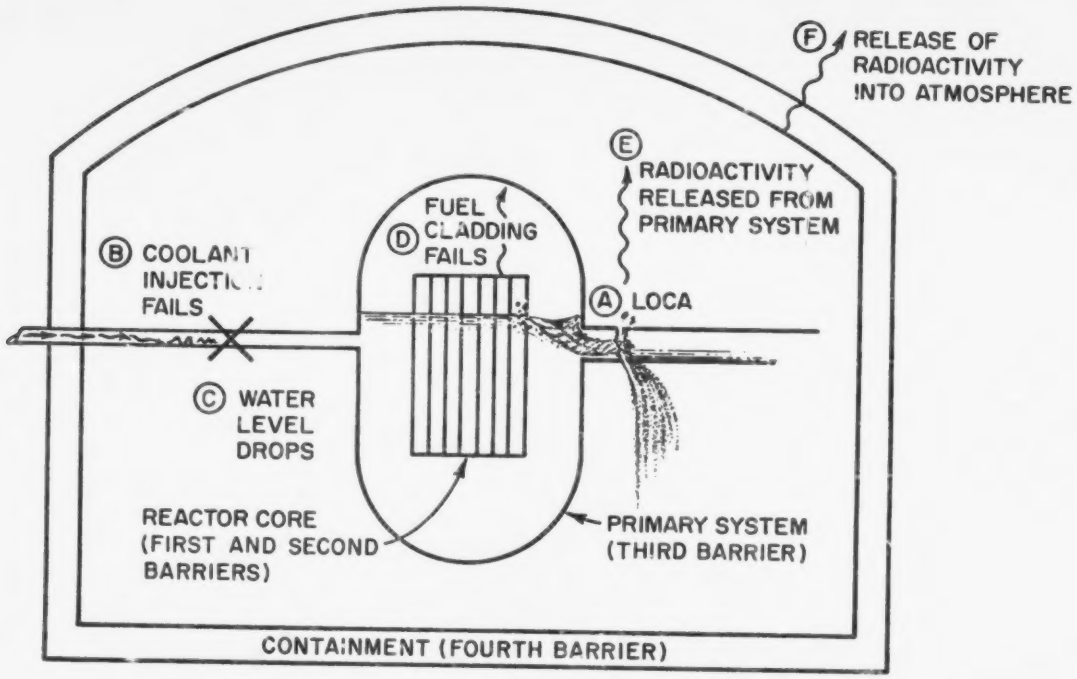
EXAMPLE OF CONTROL ROOM INDICATION BEFORE A RELEASE


(F) RELEASE OF RADIOACTIVITY INTO ATMOSPHERE
(E) RADIOACTIVITY RELEASED FROM PRIMARY SYSTEM
(D) FUEL CLADDING FAILS
(B) COOLANT INJECTION FAILS
(A) LOCA
(C) WATER LEVEL DROPS
REACTOR CORE (FIRST AND SECOND BARRIERS)
PRIMARY SYSTEM (THIRD BARRIER)
CONTAINMENT (FOURTH BARRIER)

	LOSS OF COOLANT ACCIDENT (LOCA)	EXAMPLE CONTROL ROOM INDICATORS
A	SYSTEM FAILURE – START OF ACCIDENT	PRESSURE, TEMPERATURE
B	ESF FAILURE	FLOW, TEMPERATURE
C	CSF FAILURE	CORE TEMPERATURE
D	BARRIER FAILURE	RADIATION, TEMPERATURE
E	MOVEMENT OF RADIOACTIVE MATERIAL	RADIATION
F	RELEASE TO ATMOSPHERE (CONTAINMENT LEAKAGE)	OFF-SITE DOSE RATE (e.g., AT GUARD SHACK)

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EMERGENCY ACTION LEVELS (EALs)

- Observables (e.g., control room instrument readings)
- Indicate level of emergency (classification)
  - — Activate on-site emergency organization (alert)
  - — Activate off-site organizations (site area emergency)
  - — Basis for off-site protective actions (general emergency)
- For a full range of accidents
- Most severe accidents should be classified hours (2 or more) before a major release

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EVENTS LEADING TO EVACUATION

Core Damage

↓

Potential for Early Health Effects Off Site

↓

Classify as General Emergency

↓

Immediate Recommendation of Protective Actions

↓

In General, Evacuation of Area Near Plant

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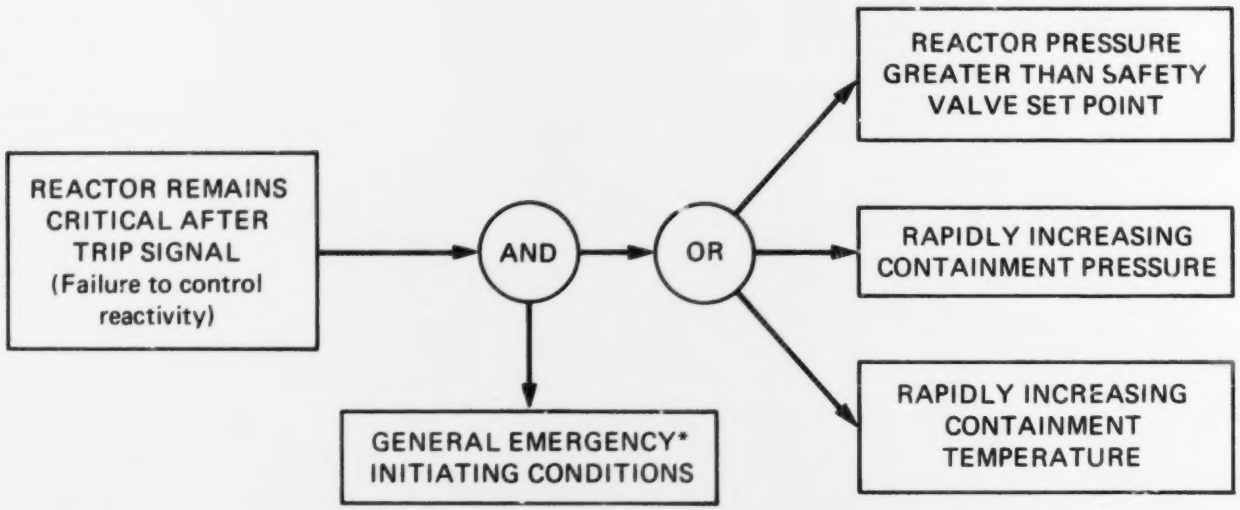
EXAMPLE OF NONRADIOLOGICAL INDICATORS OF IMMINENT SEVERE CORE DAMAGE


REACTOR REMAINS CRITICAL AFTER TRIP SIGNAL (Failure to control reactivity)
AND
OR
REACTOR PRESSURE GREATER THAN SAFETY VALVE SET POINT
RAPIDLY INCREASING CONTAINMENT PRESSURE
RAPIDLY INCREASING CONTAINMENT TEMPERATURE
GENERAL EMERGENCY* INITIATING CONDITIONS

*OBSERVABLES PERSIST FOR 15 min.

TRANSIENT REQUIRING OPERATION OF SHUTDOWN SYSTEMS WITH FAILURE TO SCRAM

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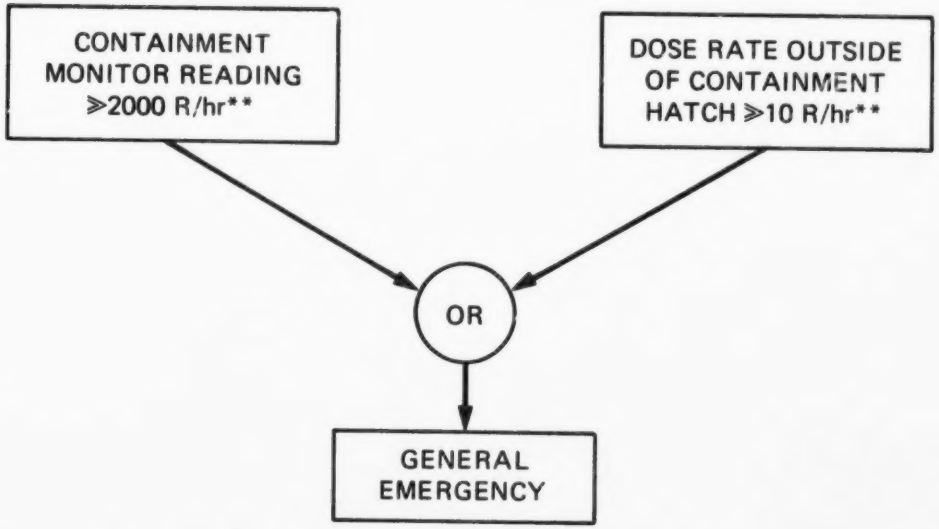
EXAMPLE OF RADIOLOGICAL INDICATORS OF SIGNIFICANT CORE DAMAGE*



*Presumed to be corroborated (i.e., not merely a faulty reading).
**Site-specific parameters – should be precalculated.

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MAJOR INDICATORS
FOR CORE DAMAGE ASSESSMENT

- Gross relationship of core damage and
  - — Radiation monitors (water, air)
  - — Water level, flow rates
  - — Thermocouple readings
  - — Pressure and pressure differences (head)
  - — Power levels (AC, DC, neutronics)

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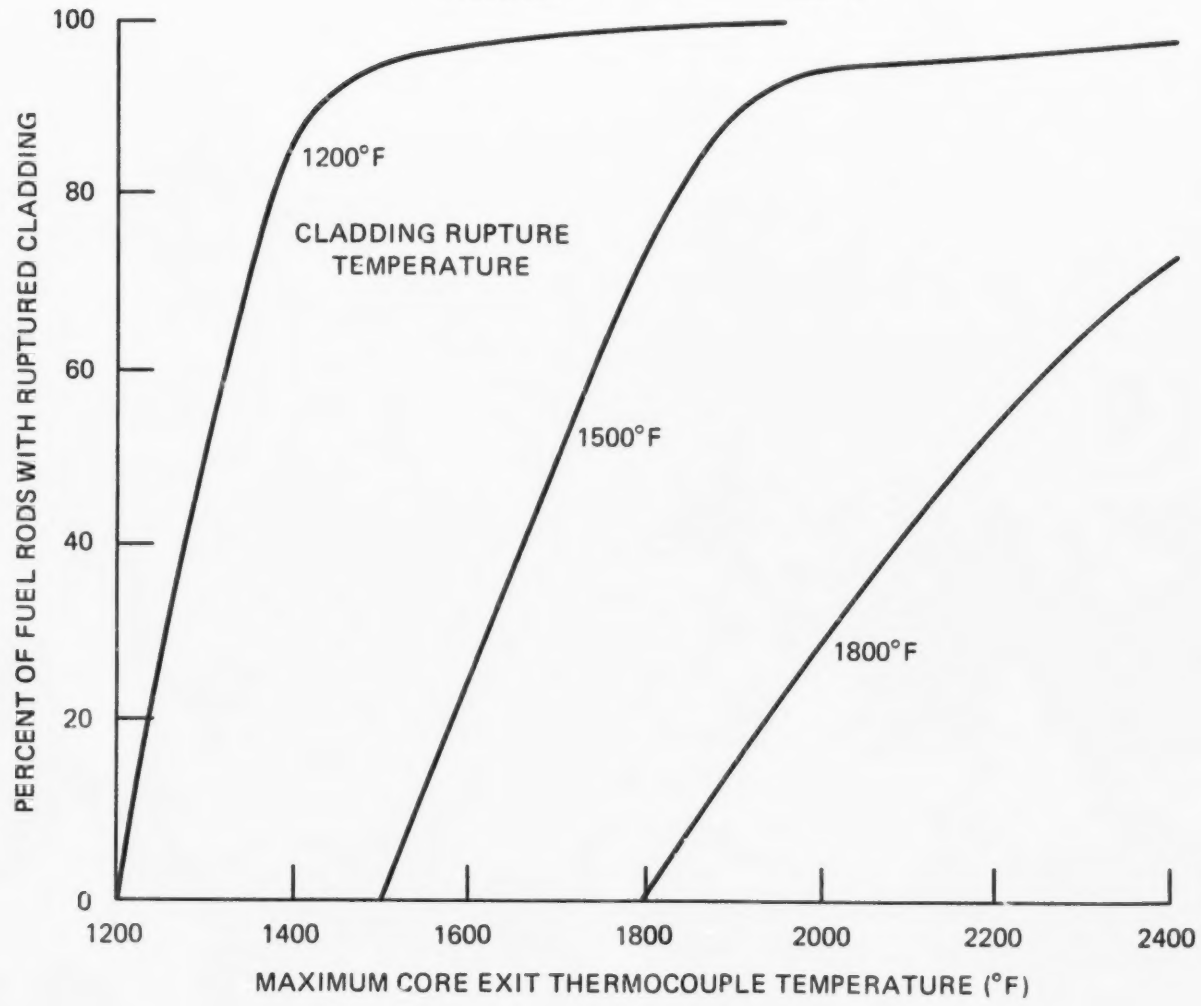
EXAMPLE OF IMPORTANCE OF TEMPERATURE AND PRESSURE MEASUREMENTS

PERCENT OF FUEL RODS WITH RUPTURED CLADDING vs
MAXIMUM CORE EXIT THERMOCOUPLE TEMPERATURE

WHEN THE PRESSURE IS	USE CURVE LABELED WITH TEMPERATURE
⩽ 100 psia	1200°F
⩽ 1200 psia	1500°F
⩽ 1650 psia	1800°F



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TYPICAL CALCULATED CONTAINMENT RADIATION ESTIMATES (R/hr)

Plant	Assumptions				Type of Reactor
	100% Gap Activity	1% Core Activity	10% Core Activity	100% Core Activity	
A	400	4,000	40,000		PWR
B	1,900	1,900	18,000	200,000	BWR
C	165,000				PWR
D		5,000	50,000	500,000	BWR
E	80,000	150	70,000	70,000	PWR
F	100,000	10,000	100,000	1,000,000	BWR

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REAL-TIME SEVERE ACCIDENT DOSE-PROJECTING PROBLEMS FOR PROTECTIVE ACTION DECISION MAKING

- May be too late for effective protective actions off site
- For a major release, evacuation must start before or soon after the release to be most effective
- Real-time dose projection would not be possible before a release
- Timing of a major release, if any, would be extremely difficult to predict in real time

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STEPS IN PREDICTING DOSE

1. Predicting quantity and of timing release (source term)
2. Predicting movement of the plume
3. Predicting dose from plume
4. Predicting dose vs distance and time

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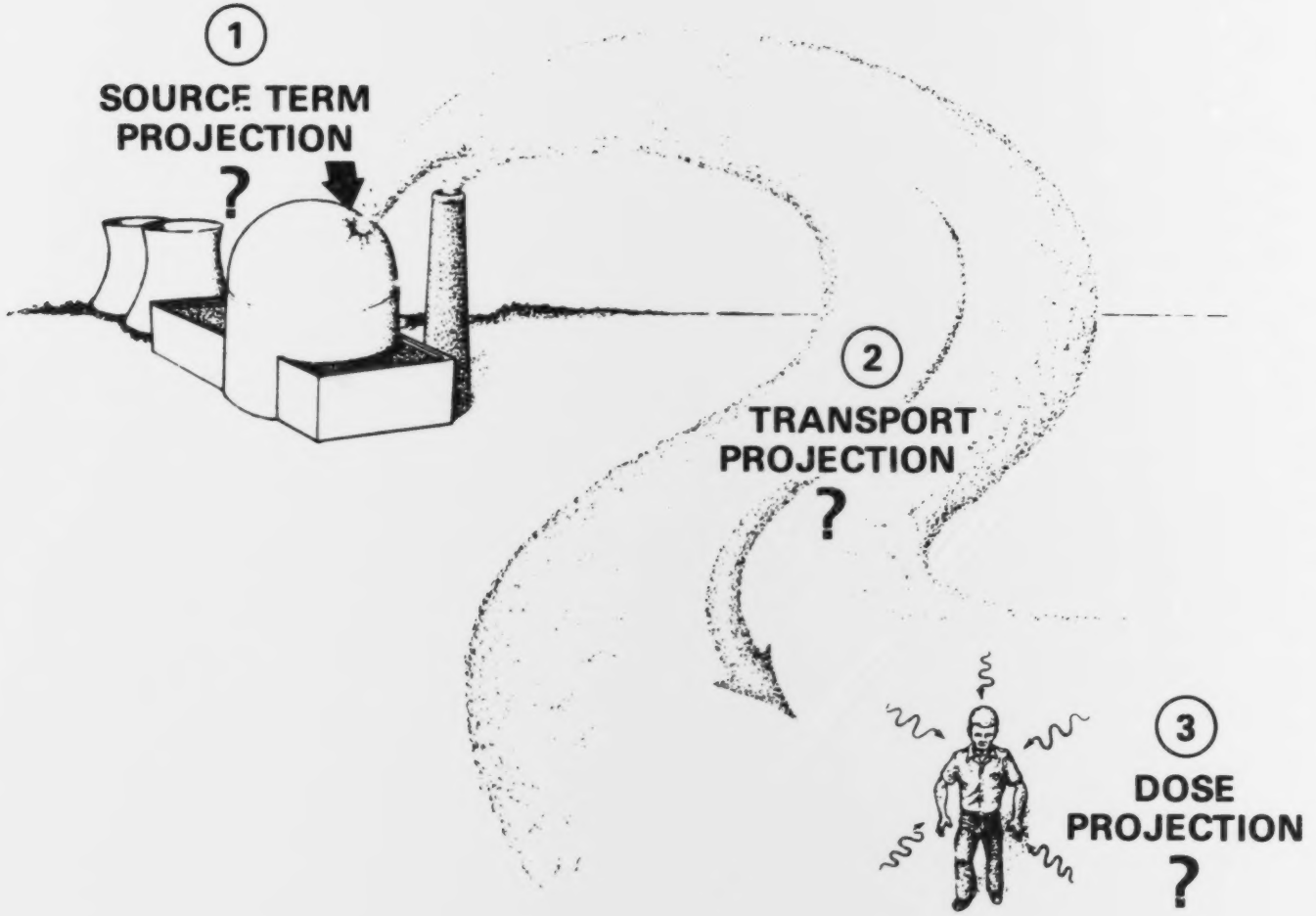
STEPS IN PREDICTING DOSE



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PREDICTING SEVERE-ACCIDENT SOURCE TERM

- Effluent radiation monitors
  - Real time: noble gas only
  - Iodine and other particulate samples: several hours
- Monitors are probably bypassed
  - Example: containment failure

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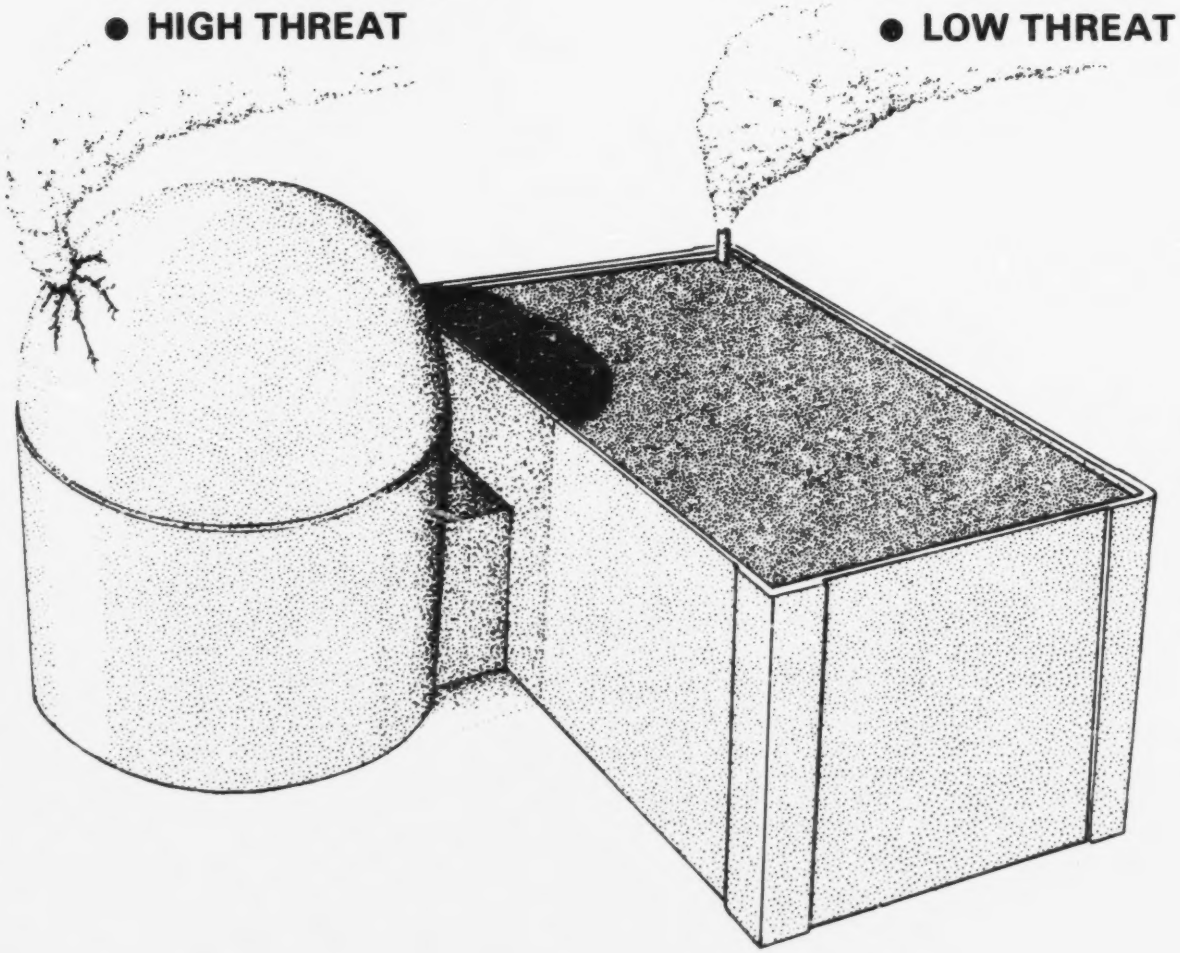
TYPES OF RELEASE

- NOT MONITORED
- HIGH THREAT

- MONITORED
- LOW THREAT



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NUREG 0956 CONCLUSIONS

- Containment response is most important
- Generalizations are inappropriate
- Reductions in risk may not be found in all cases
- There are great uncertainties in estimating source terms

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SEVERE ACCIDENT
SOURCE TERM ESTIMATION

- Early in a severe accident, it will be very difficult to predict the "source term"

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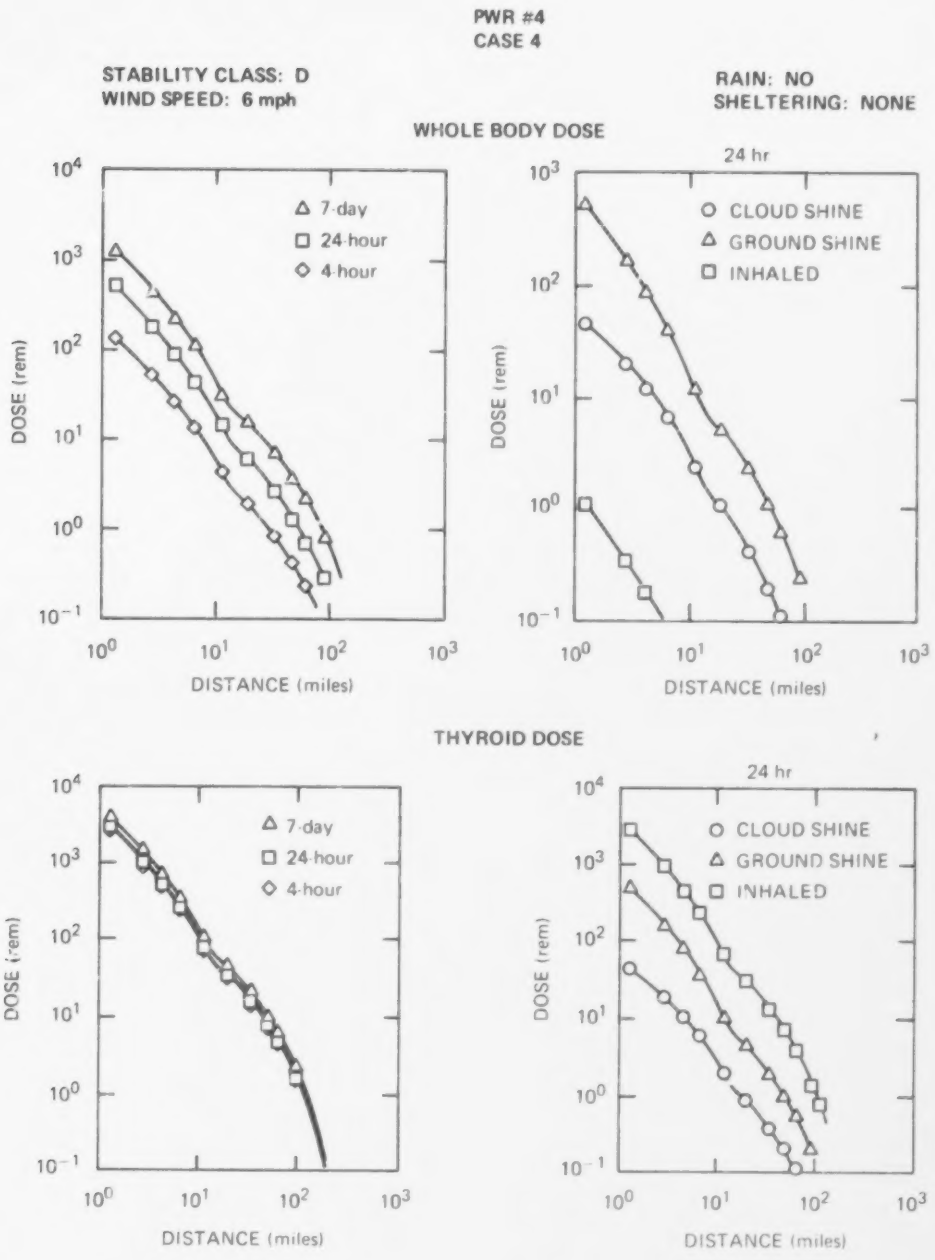
EXAMPLE FROM NUREG-1062, DOSE CALCULATIONS FOR SEVERE LWR ACCIDENT SCENARIOS



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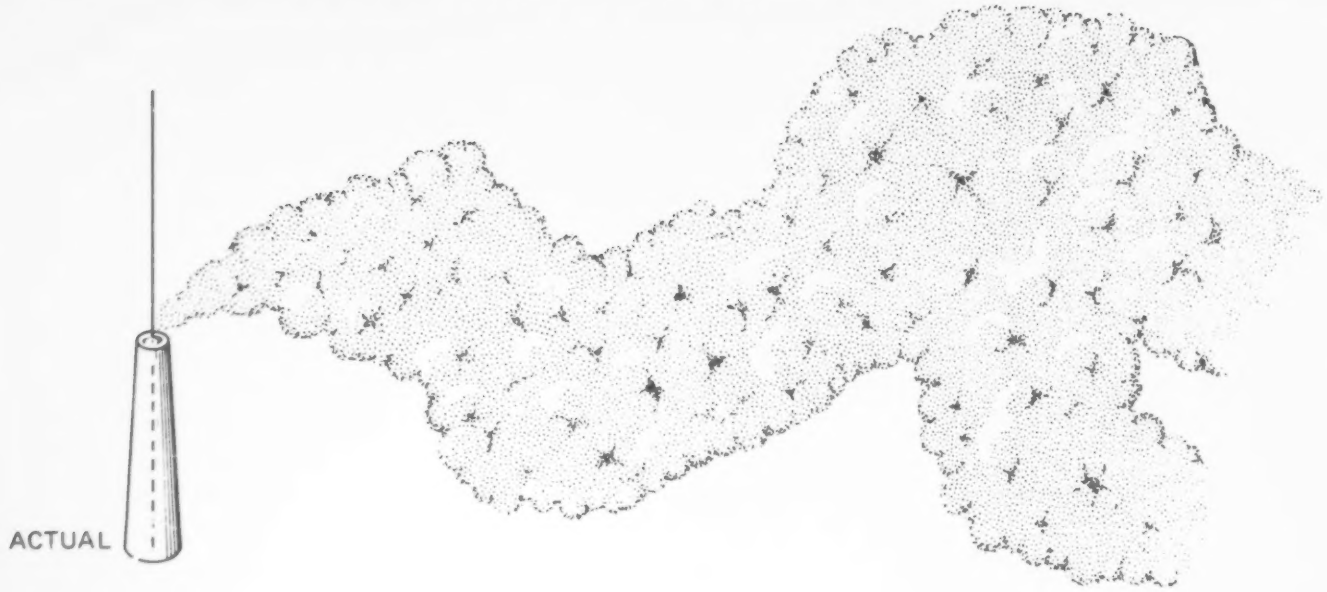
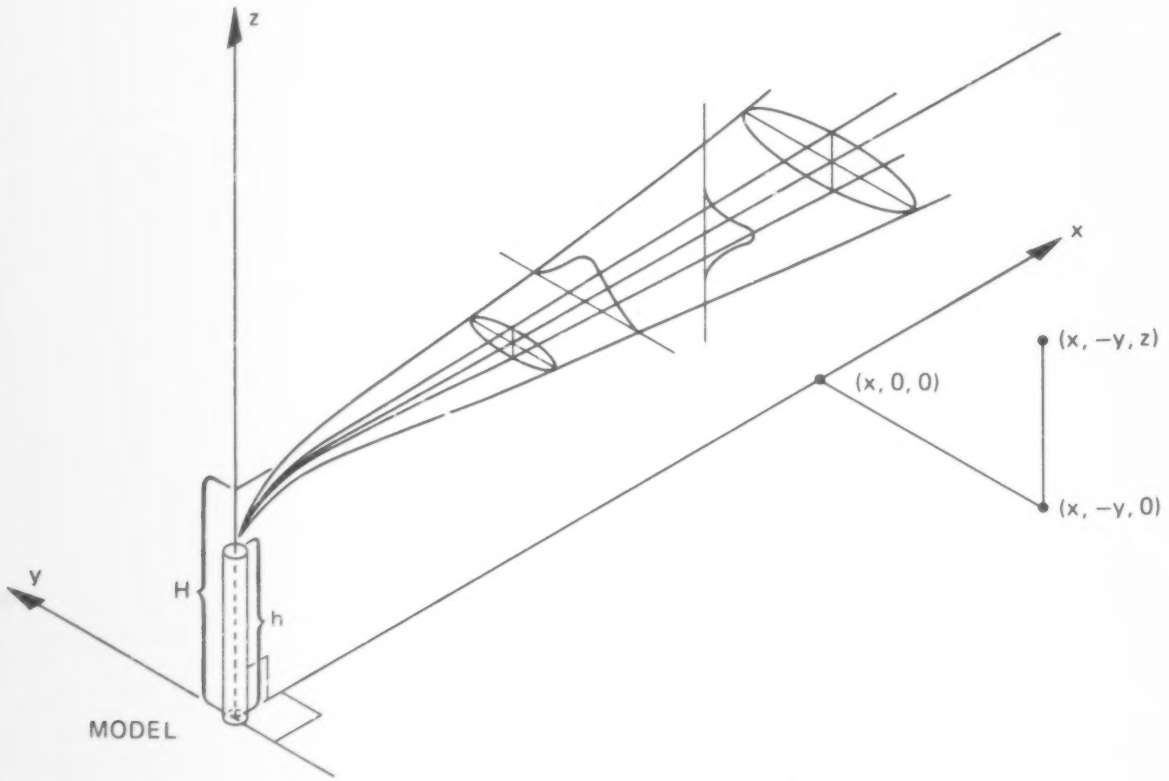
ACTUAL VS MODEL ATMOSPHERIC PLUME





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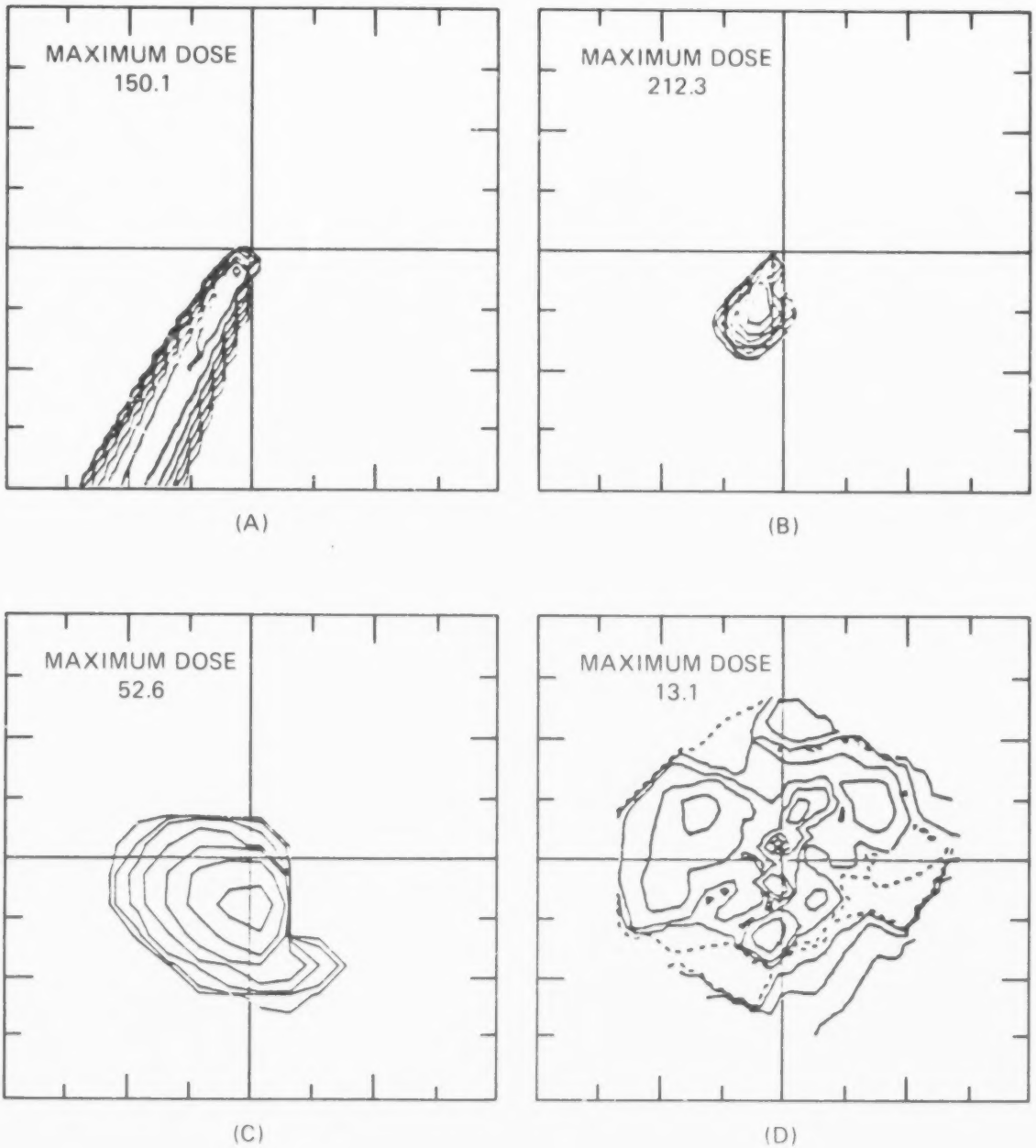
ONE-HOUR SURFACE DOSES PREDICTED BY (A) GAUSSIAN PLUME MODEL, (B) PUFF-TRAJECTORY MODEL, (C) COMPLEX NUMERICAL MODEL, AND (D) DOSES ACTUALLY OBSERVED



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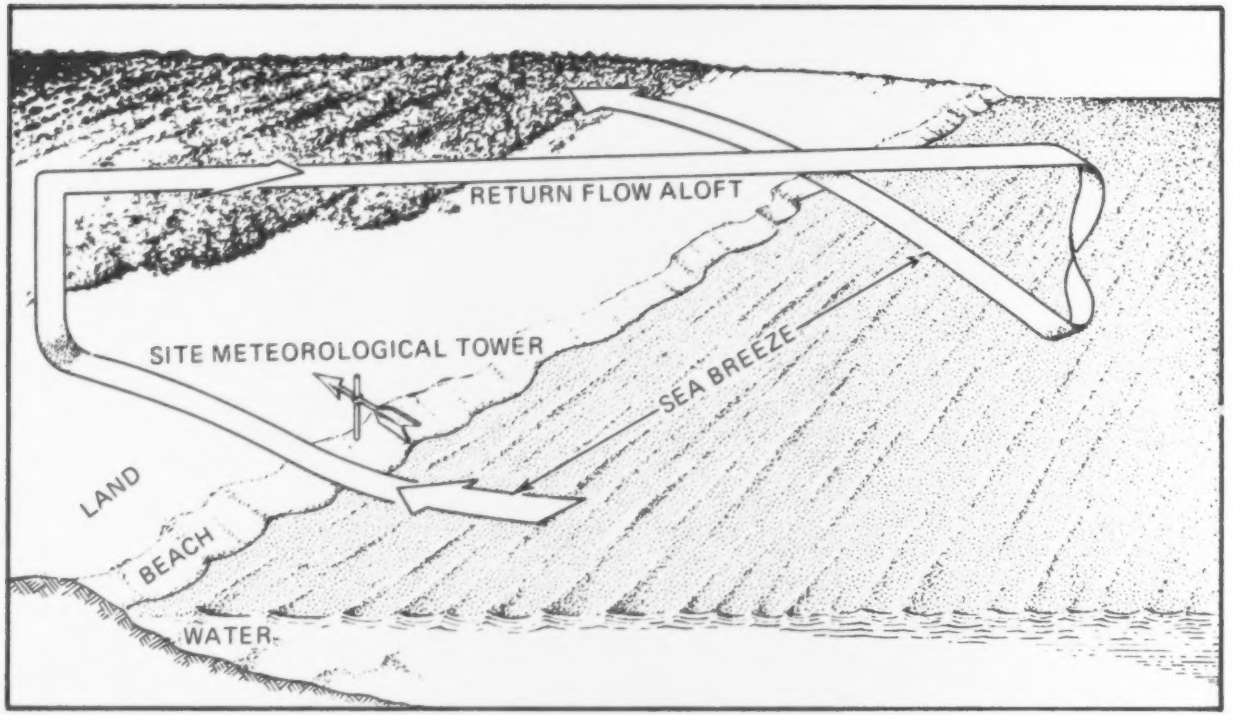
TRAJECTORY RESULTING FROM SEA BREEZE



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ORNL PHOTO 8562-85

WHAT IS STRANGE IN THIS PHOTOGRAPH?

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DIFFERENCES IN DOSE MODEL RESULTS

- The results of various dose models differ because of
  - Different methods
  - Different assumptions
  - Different objectives
- Expect a 10- to 100-fold spread
- Knowledge of local meteorological trends and conditions is most important

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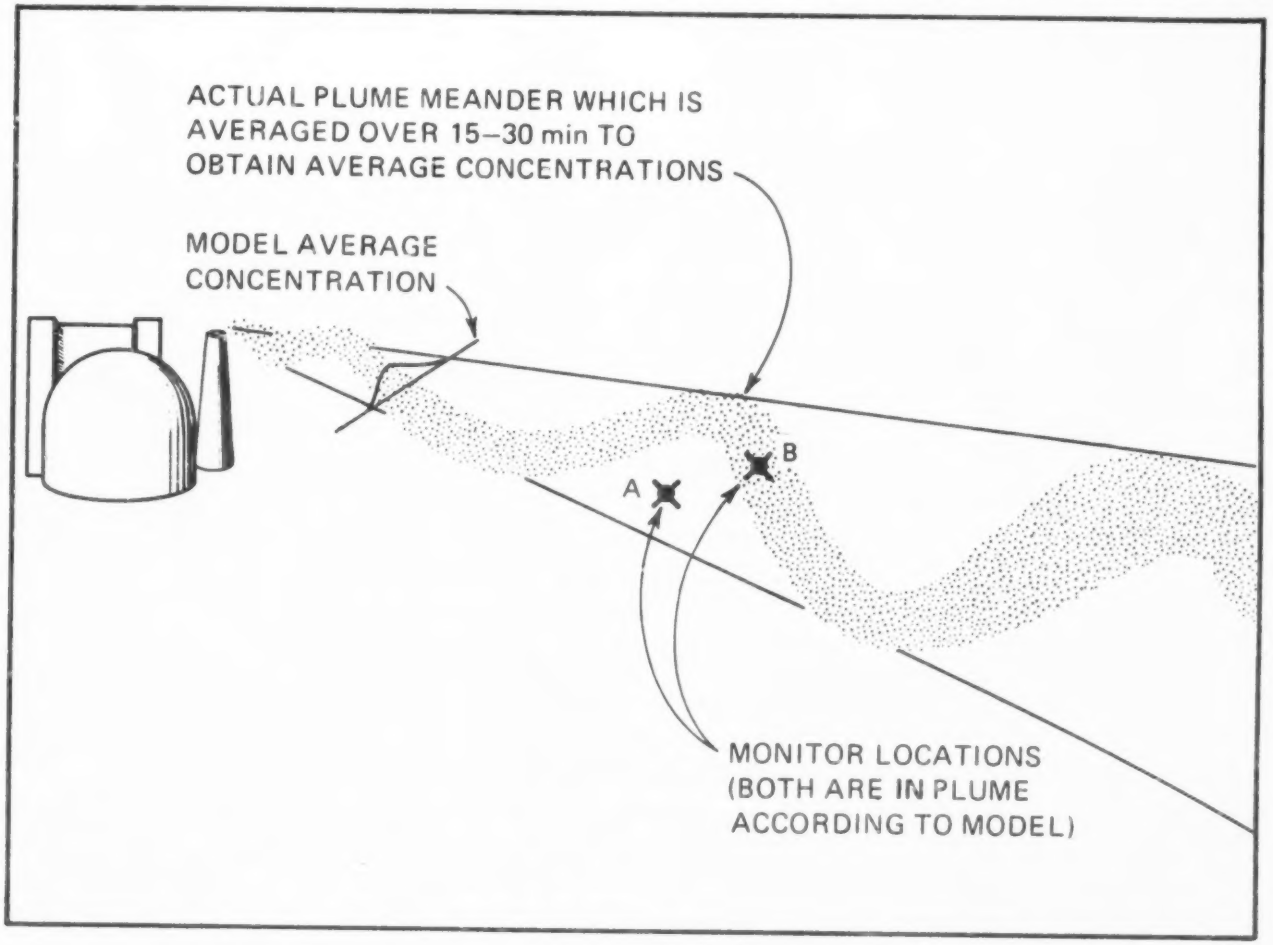
RELATIONSHIP BETWEEN ACTUAL PLUME AND MODEL PROJECTIONS



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ESTIMATED RANGE OF UNCERTAINTY BETWEEN PROJECTED AND ACTUAL OFF-SITE DOSE FOR A SEVERE ACCIDENT (CORE MELT)

Element	Uncertainty Factor[a]		
	At Best	Most Likely	Near Worst
Source term (event and sequence)	5	100–1,000	100,000
Dispersion			
Diffusion (concentration)	2	5	10
Transport (direction)	22°	45°	180°
Transport (rate)	1	2	10 (low wind speed)
Dosimetry	3	4	5
Overall (dose and direction)	10, 22°	100–10,000, 45°	100,000, 180°

[a]These estimates are for an averaged dose at a location (e.g., 15–30 min), not for a specific or single monitor reading.

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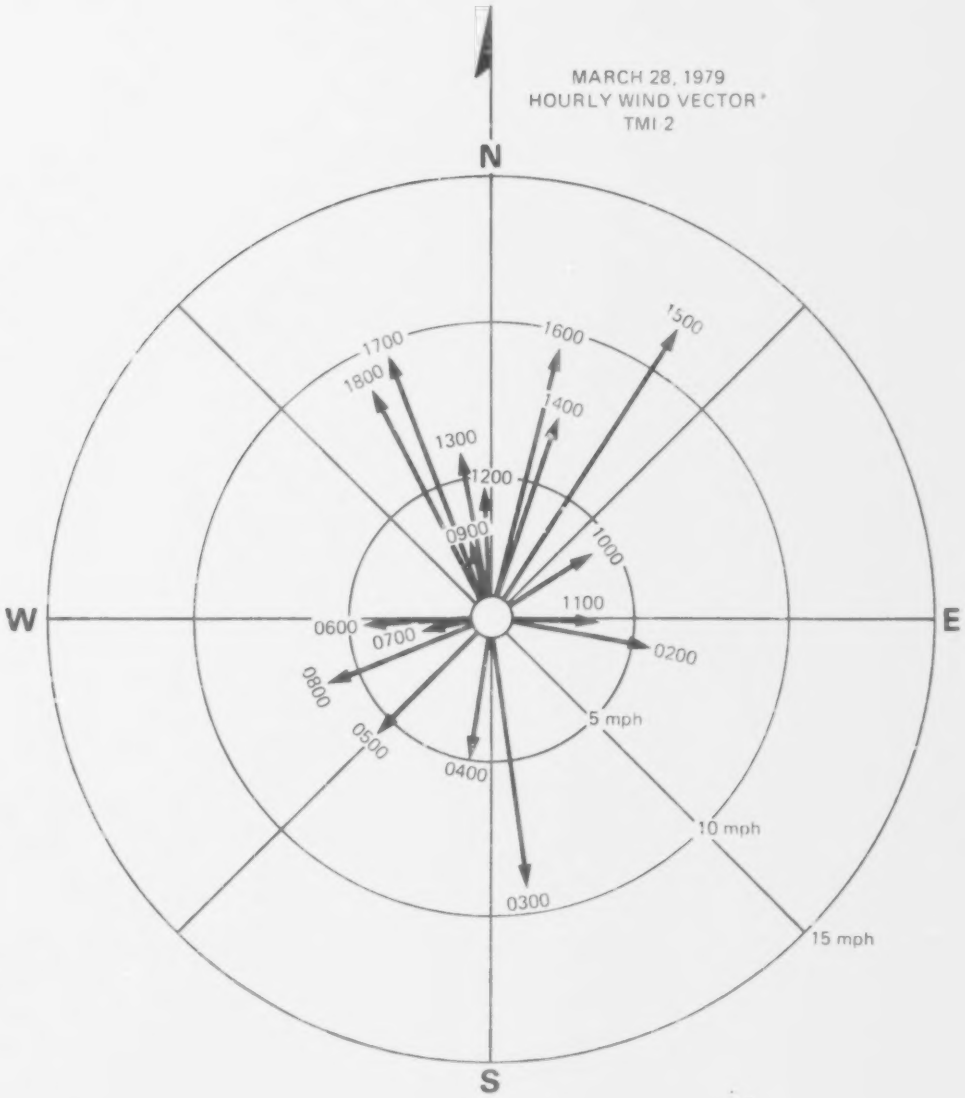
HOURLY WIND VECTOR AT TMI-2 ON MARCH 28, 1979



*Arrows indicate direction toward which the on site wind was blowing at the local time indicated. Circles represent varying wind speeds.

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TMI-2 PLANT CONDITIONS AT 9:00 A.M.

- Clear indications of core damage
  - Containment monitor – 6000 R/hr
  - Thermocouple – 2000°F
    (Cladding failure at 1200°F)
- Plant beyond analyzed conditions
- Large amount of radioactive material in containment (only 1 barrier remains)

Problems

- No protective action taken because projected doses off site did not exceed PAGs
- Action not recommended on plant (core) conditions
- 300° wind shift in 3 hr

Findings

- Initiate early protective actions off site in all directions
- Initiate early protective actions off site on detection of core damage

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THE ROLE OF DOSE ASSESSMENT DURING VARIOUS PHASES OF NUCLEAR POWER PLANT ACCIDENTS

- Initial phase of general emergency
  - Determine areas to be monitored first
  - Compare consequences of various plant response options
  - Determine maximum distance requiring further protective actions
- Site area emergency and intermediate phase of a general emergency
  - Assess additional areas for implementation of protective actions
  - Provide feedback on source term based on monitoring results

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FIELD-MONITORING ROLE IN PROTECTIVE ACTION DECISION MAKING AFTER A MAJOR RELEASE

- After major release
- Identify areas in which early protective action decisions should be modified
- Can provide indication of actual off-site radiological impacts

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FIELD-MONITORING PROBLEMS

1. Large variations–requires a large number of monitoring results to characterize a release
2. May not be in plume–over teams–meanders or loops around team
3. Concentrations could change quickly
4. Release could be very complex

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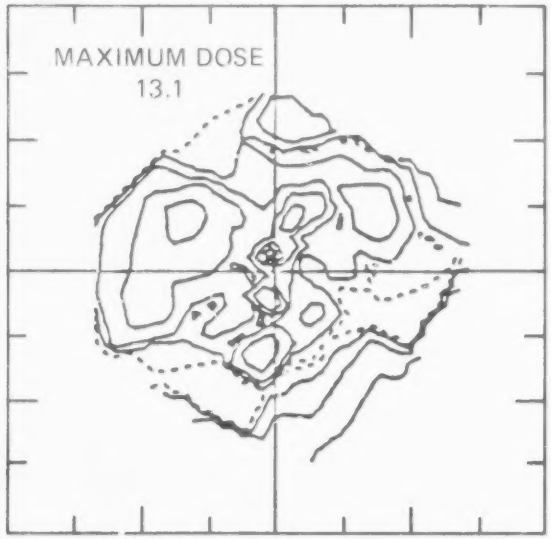
ONE-HOUR SURFACE DOSES PREDICTED BY DOSES ACTUALLY OBSERVED



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MAJOR POINTS

- Early, predetermined protective actions off site unwarranted for less than general emergencies
- Core can be kept cool by using decay heat to boil water (1000 gal/min)
- Release of only small fraction of core inventory could result in health effects
- Core damage and fast, direct release required for severe health effects
- Can't miss core damage–can't predict release
- Early off-site protective actions based on core damage–don't wait for dose assessments–use predetermined EALs
- Core damage–evacuate near the site if possible
- Control room very busy during an emergency

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MAJOR POINTS (continued)

- During a major accident, accurate early dose projection may be impossible
- Early protection could be highly uncertain with the major problem being source term estimates
- Considerable differences will occur between field measurements and model predictions
- Considerable differences can exist between dose projection models even if same inputs are used
- Dose assessment has a secondary role during early stages of a major accident–use dose assessment after initial early actions are taken based on core conditions
- Take protective actions in all directions near the plant, when warranted at all
- Field monitoring is most accurate indication of extent of off-site radiological impacts–but early results should be used with caution

Slide 62 (Continued)

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NRC FORM 335 (2 84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

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5 AUTHOR(S)

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12 SUPPLEMENTARY NOTES

13 ABSTRACT (200 words or less)

This is one in a series of volumes that collectively provide for the U.S. Nuclear Regulatory Commission (NRC) emergency response personnel the necessary background information for an adequate response to severe reactor accidents. The volumes in the series are:

- Volume 1 -- Overview and Summary of Major Points
- Volume 2 -- Severe Reactor Accident Overview
- Volume 3 -- Response of Licensee and State and Local Officials
- Volume 4 -- Public Protective Actions -- Predetermined Criteria and Initial Actions
- Volume 5 -- U.S. Nuclear Regulatory Commission Response

Each volume serves, respectively, as the text for a course of instruction in a series of courses for NRC response personnel. These materials do not provide guidance or license requirements for NRC licensees or state or local response organizations. Each volume is accompanied by an appendix of slides that can be used to present this material. The slides are called out in the text.

14 DOCUMENT ANALYSIS - a KEYWORDS/DESCRIPTORS

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power reactor accidents
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b IDENTIFIERS/OPEN ENDED TERMS

15 AVAILABILITY STATEMENT

16 SECURITY CLASSIFICATION (This page)

(This report)

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